4.3.5 Implementation Alternative 5: Wet Storage Technology for New Construction

Wet storage technology for new construction was considered instead of the dry storage technology contained in the basic implementation of Management Alternative 1, for all five potential foreign research reactor spent nuclear fuel management sites. The impacts during marine transport, port activities, and ground transport would be the same as in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the analysis examined environmental topics including land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational health and safety, noise, utilities and energy, and waste management.

The means by which this alternative would be implemented at each management site are presented in Sections 2.6.5.3.1 through 2.6.5.3.5. The environmental impact analysis assumes that a new wet storage facility, which is described in Section 2.6.5.1.2, would be constructed at the sites to receive and store foreign research reactor spent nuclear fuel after the Phase 1 period. At the Savannah River Site, the alternative could also be implemented at the Barnwell Nuclear Fuels Plant (BNFP) and at the Hanford Site by the addition of facilities to the WNP-4 Spray Pond. These facilities are described in Appendix F, Section F.3. The analysis parallels in all respects the impact analysis performed for the new dry storage facility of the basic implementation of Management Alternative 1. It is presented in detail in Appendix F, Section F.4, with methodology and assumptions for radiological impacts given in Sections F.5 and F.6.

As in the basic implementation of Management Alternative 1, the analysis showed that this implementation alternative would not cause any major environmental impacts. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

4.3.5.1 Occupational and Public Health and Safety

As in the basic implementation of Management Alternative 1 (see Section 4.2.4.1) radiological exposures are presented as emissions-related impacts, handling-related impacts, and accident-related impacts.

Impacts to the Public of Incident-Free Management Site Activities

Table 4-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at each Phase 2 site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

The highest estimated Phase 1 public MEI and population risks for this alternative are identical to those for the basic implementation of Management Alternative 1. All possible Phase 1 MEI risks are lower than the highest estimated Phase 2 MEI risk in the next paragraph, so they will drop out. The highest Phase 1 component of the population risk is 0.00014 LCF in the basic implementation.

Among all the potential Phase 2 foreign research reactor spent nuclear fuel management sites, the maximum annual dose to the public from emissions is 0.06 mrem per year and 0.06 person-rem per year at the Oak Ridge Reservation for the MEI dose and the population dose, respectively. If it is assumed that receipt of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation could take place over a period of 3 years, the total MEI dose would be 0.18 mrem and the total population dose would be 0.18 person-rem. If it is further assumed that storage will continue for 30 years after the beginning of the receipt period, the total MEI dose from storage would be 1.4×10^{-5} mrem and the total population dose from storage would be 1.5×10^{-5} person-rem. The risks due to receipt and unloading would be much

Table 4-39 Annual Public Impacts for Receipt and Management of Foreign Research Reactor Spent Nuclear Fuel Under Implementation Alternative 5 (Wet Storage)

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Savannah River Site				
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.0000060
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
• New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at:				
• New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}
Nevada Test Site				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at:				
• New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

higher than those due to storage, so the maximum risk would be 0.18 mrem for the MEI and the sum of population doses would be 0.18 person-rem. The associated probabilities for incurring one LCF would be 9×10^{-8} LCF for the Phase 2 MEI risk and 0.00009 LCF for the Phase 2 population risk.

The maximum of the Phase 1 and Phase 2 incident-free public MEI risks is 9×10^{-8} LCF for this alternative. The sum of the Phase 1 and Phase 2 incident-free public population risks is 0.00023 LCF.

Impacts to Workers of Incident-Free Management Site Activities

As in the basic implementation of Management Alternative 1, workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the site or transferring foreign research reactor spent nuclear fuel from one facility to another within the site. The methodology and assumptions for the analysis of this implementation alternative parallel that for the basic implementation of Management Alternative 1 as presented in Section 4.2.4.1 and Appendix F, Section F.5.

Table 4-40 presents the collective doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at the site.

Table 4-40 Handling-Related Impacts to Workers at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
Savannah River Site		
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP[a]	310	0.12
Idaho National Engineering Laboratory		
Phase 1: IFSF/CPP-749	257	0.10
Phases 1 and 2: New Wet Storage Facility	367	0.15
Phase 1: FAST	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Hanford Site		
Phase 2: New Wet Storage Facility or WNP-4 Spray Pond	109	0.04
Oak Ridge Reservation		
Phase 2: New Wet Storage Facility	109	0.04
Nevada Test Site		
Phase 2: New Wet Storage Facility	109	0.04

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

As seen from Table 4-40, the maximum total collective dose to workers handling foreign research reactor spent nuclear fuel at a single site would be 367 person-rem for the case analyzed at the Idaho National Engineering Laboratory, which assumes that all foreign research reactor spent nuclear fuel is in dry storage during Phase 1 and is transferred to a new wet storage facility for Phase 2. The associated probability for one LCF among the working crew would be 0.15. The highest dose to working crews for both phases in more than one site is 366 person-rem: 109 person-rem at one of the three Phase 2 sites plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.15.

Accident-Related Impacts

The accident scenarios analyzed for this implementation alternative are the same as those analyzed for the basic implementation of Management Alternative 1.

Table 4-41 presents the frequency and consequences of the accidents analyzed for each management site for this implementation alternative. Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. Table 4-42 presents the annual risk estimates for wet storage.

The highest MEI or NPAI risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1 (2.6×10^{-6} LCF). The highest annual MEI or NPAI risk for Phase 2 would be 0.000005 LCF per year, which is the annual risk to the NPAI from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this MEI/NPAI risk would

Table 4-41 Frequency and Consequences of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Savannah River Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	17	0.0000085	9.5	0.0000048	370	0.19	1,600	0.00064
• Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
BNFP									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	9×10^{-9}	0.00099	5×10^{-10}	0.028	0.000014	0.00080	3.2×10^{-10}
• Accidental Criticality[a]	0.0031	80	0.000040	75	0.000038	44	0.022	75	0.000030
• Aircraft Crash	1×10^{-6}	92	0.000046	31	0.000016	23	0.012	70	0.000028
Idaho National Engineering Laboratory									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1,800	0.00072
• Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016
Hanford Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	6.5×10^{-8}	0.0033	1.7×10^{-9}	1.6	0.00080	0.25	1.0×10^{-7}
• Accidental Criticality	0.0031	64	0.000032	14	0.000007	740	0.37	3,600	0.0014
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
WNP-4 Spray Pond									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.15	7.5×10^{-8}	0.0033	1.7×10^{-9}	1.3	0.00065	0.00024	9.6×10^{-11}
• Accidental Criticality[a]	0.0031	97	0.000049	76	0.000038	620	0.31	120	0.000048
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Oak Ridge Reservation									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	3.6×10^{-7}	0.20	1.0×10^{-7}	16	0.0080	0.68	2.7×10^{-7}
• Accidental Criticality	0.0031	1,500	0.00075	3,300	0.0017	1,400	0.70	6,800	0.0027
• Aircraft Crash	1×10^{-6}	380	0.00019	600	0.00030	2,900	1.5	1,900	0.00076
Nevada Test Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	2.7×10^{-8}	0.0016	8×10^{-10}	0.33	0.00017	0.10	4.0×10^{-8}
• Accidental Criticality	0.0031	88	0.000044	15	0.0000075	54	0.027	1,300	0.00052
• Aircraft Crash	1×10^{-6}	29	0.000015	4.2	0.0000021	61	0.031	290	0.00012

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

Table 4-42 Annual Risks of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
	Savannah River Site			
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-7}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}
	Idaho National Engineering Laboratory			
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}
	Hanford Site			
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[b]	NA	NA	NA	NA
WNP-4 Spray Pond				
• Spent Nuclear Fuel Assembly Breach[a]	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality[a]	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[b]	NA	NA	NA	NA
	Oak Ridge Reservation			
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}
	Nevada Test Site			
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

be 0.00015 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.00015 LCF for the maximum MEI risk due to accidents.

The highest population risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1, 0.096 LCF. The highest annual population risk for Phase 2 would be 0.0022 LCF per year, which is the annual risk to the public from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the

Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this population risk would be 0.066 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Adding the Phase 1 and Phase 2 population risks yields 0.16 LCF for the total population risk due to accidents.

4.3.5.2 Topics Not Discussed in Detail

Nonradiological impacts associated with the wet storage implementation alternative are similar to those for dry storage considered in the basic implementation of Management Alternative 1. They are discussed in detail in Appendix F, Section F.4.

Impacts at each management site typically associated with construction activities such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, air quality, ecology, and noise are similar because: (1) both dry and wet storage facilities could be constructed at the same locations at each site; and (2) both facilities are approximately the same size. As indicated in Section 2.6.5.1, the construction of the wet storage facility would disturb approximately 2.8 ha (7 acres) of land while the construction of the dry storage facility would disturb 3.6 to 4.5 ha (9 to 11 acres). Specifically for the Savannah River Site, if the wet storage alternative is implemented using the BNFP facility there would be no impacts associated with construction activities.

Impacts at each management site typically associated with the operation of the facilities such as air quality, water quality, socioeconomics, utilities, and waste generation are also very similar as indicated in Section 2.6.5.1. The only notable difference is indicated in water use. The wet storage facility would use 1.5 million liters (409,000 gal) per year during the storage mode of the operation (over 30 years) compared to 0.9 million liters (238,000 gal) per year used by the dry storage facility over the same period. This difference, however, is small compared to typical water consumption rates at the sites: 1.14 billion liters (300 million gal) per year at the Nevada Test Site to 88 billion liters (23.2 billion gal) per year at the Savannah River Site.

4.3.5.3 Summary of the Impacts of Implementation Alternative 5

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-43 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-43 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-43 Maximum Estimated Radiological Health Impacts of Implementation Alternative 5 (Wet Storage)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.071
Accidents	1.4×10^{-11}	0.00028	---
Site Activities			
Incident-Free	0.026	0.00023	0.15
Accidents	0.00015	0.16	---
Highest Individual Risk			
Incident-Free	0.026	---	---
Accidents	0.00015	---	---
Total Population Risk			
Incident-Free	---	0.22	0.27
Accidents	---	0.16	---

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 9×10^{-8} LCF.

The highest estimated accident MEI risk is 0.00015 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than two in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-43, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.27 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 27 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-43. There is about a 14 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.6 Implementation Alternative 6: Near Term Chemical Separation in the United States

As discussed in Section 2.2.2.6, this implementation alternative involves conventional chemical separation in existing facilities at either the Savannah River Site or the Idaho National Engineering Laboratory. The facilities at the Savannah River Site are limited to chemically separating the aluminum-based foreign research reactor spent nuclear fuel. After some upgrading, the facilities at the Idaho National Engineering Laboratory would have the capability to chemically separate all the foreign research reactor spent nuclear fuel.

4.3.6.1 Implications of Chemical Separation for U.S. Nonproliferation Policy

As a matter of policy, the United States does not currently engage in reprocessing or chemical separation to extract plutonium for civilian or military purposes. U.S. policy is also not to encourage the civilian use of plutonium and to explore means to limit the stockpiling of plutonium from civil nuclear programs. This alternative nonetheless considers scenarios whereby the United States might engage in future chemical separation of foreign research reactor spent nuclear fuel. If a decision were made pursuant to this EIS to chemically separate some or all of the foreign research reactor spent nuclear fuel, the limited amount of plutonium in the spent fuel would not be separated. Rather it would be left in, and disposed of with, the high-level radioactive wastes produced during the chemical separation operation.

Two alternatives are evaluated for handling the highly enriched uranium in the spent fuel, either to blend it down to low enriched uranium (the preferred alternative, if any chemical separation is undertaken), or to separate it as HEU and place it in safe, secure storage. Chemical separation of foreign research reactor spent nuclear fuel, with blending down of the separated uranium, would, in fact, result in a reduction in the amount of HEU – a major goal of the U.S. Nuclear Weapons Nonproliferation Policy announced in September 1993. Despite this fact, there is a concern that other states may perceive only that the U.S. has restarted reprocessing.

For example, the potential exists that other states (e.g., Iran), might use the restart of reprocessing in the United States as an excuse to continue current programs or begin new ones – activities that would run counter to U.S. nuclear weapons nonproliferation interests. The implications in North Korea, where the United States has been actively working to create a nonreprocessing zone, as well as in other states, could complicate current U.S. nonproliferation activities.

4.3.6.2 General Assumptions and Analytic Approach

Potential impacts at the Savannah River Site and the Idaho National Engineering Laboratory were estimated separately. The impacts due to chemical separation and associated onsite activities would be in addition to those due to marine transport, port activities, and ground transport.

As discussed in Section 2.2.2.6, DOE and the Department of State have analyzed four possible chemical separation subalternatives under this implementation alternative. These four subalternatives, with spent nuclear fuel amounts and estimated facility run durations are:

	Amount (MTHM)	***Duration (Years)***
Savannah River Site (only aluminum-based spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	18.2	13
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	51	13
Idaho National Engineering Laboratory (aluminum-based and TRIGA spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	19.2	12
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	65	12

The duration of chemical separation operations dedicated to foreign research reactor spent nuclear fuel is driven by the rate of foreign research reactor spent nuclear fuel receipt at the Savannah River Site or the Idaho National Engineering Laboratory. The facility run durations at Savannah River Site are both up to 13 years, whether the facilities would be chemically separating only the 18.2 MTHM of foreign research reactor spent nuclear fuel or the 51 MTHM of spent nuclear fuel. Because the additional spent nuclear fuel would be chemically separated at the same time as the foreign research reactor spent nuclear fuel in a parallel process, only the combined impacts will be used to determine the risks associated with the overall operations. There are other nuclear materials, such as the Mark-31 targets currently stored at the Savannah River Site, which could also be chemically separated. These nuclear materials are not included in this implementation alternative, but they are covered under cumulative impacts. The impacts of running the facilities are based on conservative assumptions regarding incident-free annual emissions and possible accident releases which cover this range of throughputs.

The facility run durations at the Idaho National Engineering Laboratory are estimated to be up to 12 years. Furthermore, the same type of conservative assumptions regarding incident-free emissions and accidental releases are applied to calculate the environmental impacts.

As discussed in Section 2.2.2.6, the implementation component of uranium disposition has policy implications. The separated LEU could be returned to the commercial sector for reuse as reactor fuel. The HEU could be blended down to LEU or it could be processed directly to an oxide and stored. If a decision is made to chemically separate this spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

4.3.6.3 Marine Transport Impacts

The marine transport impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.1.

4.3.6.4 Port Activities Impacts

The port activities impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.2.

4.3.6.5 Ground Transport Impacts

The impacts due to ground transport of foreign research reactor spent nuclear fuel in this implementation alternative would be slightly lower than those of the basic implementation of Management Alternative 1, because the Phase 2 intersite shipments would not occur.

If the aluminum-based foreign research reactor spent nuclear fuel were chemically separated at the Savannah River Site it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period. The TRIGA foreign research reactor spent nuclear fuel would be transported to either of the two sites for management and it would not be transported again for the duration of the 40-year program period.

Similarly, if all the foreign research reactor spent nuclear fuel were chemically separated at the Idaho National Engineering Laboratory, it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period.

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of management sites that created varying cask shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCF for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCF from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.3×10^{-11} LCF.

4.3.6.6 Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

DOE and the Department of State evaluated near term chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory for five key types of impacts: (1) Socioeconomics, (2) Air Quality, (3) Water Quality, (4) Occupational and Public Health and Safety, and (5) Waste Management. The other impacts are all the same as those described in the basic implementation of Management Alternative 1. The analytic approach was to use the results published in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the Interim Management of Nuclear Materials Final EIS (DOE, 1995a) whenever possible. Usually, these results can be adopted directly.

4.3.6.6.1 Socioeconomics

Savannah River Site

The chemical separation facilities at the Savannah River Site were last operated in 1992. The facilities are in a warm standby condition and are currently fully staffed. Use of these facilities would not have a notable net impact upon employment or the regional economy.

Idaho National Engineering Laboratory

The chemical separation facilities at the Idaho National Engineering Laboratory were last operated in 1990 and are currently in the process of being cleaned out in preparation for decommissioning. Some staff would need to be added eventually, but the use of these facilities would not have a notable net impact upon employment or the regional economy.

4.3.6.6.2 Air Quality

Savannah River Site

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Savannah River Site in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions would be small increases over baseline site-wide totals and within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has analyzed the expected airborne radiological emissions from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These radiological emissions are presented in Table 4-44 (Grainger, 1995). The health effects from these airborne emissions are discussed in Section 4.3.6.6.4 below.

Table 4-44 Annual Incident-Free Airborne Radiological Emissions at the Savannah River Site that Contribute to the Offsite Dose[a]

Element	*Ci/yr*
Tritium	57.8
Cesium-134	0.002
Cesium-137	0.12
Curium-242/244	0.12
Cerium-144	0.0059
Americium-241	0.016
Cobalt-60	0.000000053
Plutonium-238	0.078
Plutonium-239	0.020
Strontium-89/90	0.17
Iodine-131	0.0053
Uranium-235/238	0.039
Antimony-125	0.018
Ruthenium-106	0.20

[a] *Krypton-85 would be released at an estimated rate of 120,000 Ci/yr*

Source: Grainger, 1995

Krypton-85 emissions are not included in Table 4-44 because these releases are not normally measured or calculated. The health effects resulting from krypton-85 releases are very low compared to those resulting from other isotopes that are being measured. Krypton is an inert gas with no affinity for biological systems, so it does not adhere to the lungs if inhaled. The radioactive isotope of krypton would cause such a low level of harm to the population near the Savannah River Site because it remains in the human body for only very brief periods of time. The total amount of krypton-85 that would be contained in all of the aluminum-based foreign research reactor spent nuclear fuel is conservatively estimated to be 1.5 x 10^6 curies. Assuming this is released gradually over the 12-year reprocessing period, the annual emission rate would be 1.2 x 10^5 curies per year.

Idaho National Engineering Laboratory

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Idaho National Engineering Laboratory in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions are within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has also analyzed the expected radiological emissions from the Idaho National Engineering Laboratory chemical separations facilities in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These are presented in Table 4-45. The radiological emission rates were estimated using conservative engineering calculations based on knowledge of the proposed activity. These emission rates are representative of emissions that could occur during Implementation Alternative 6 at the Idaho National Engineering Laboratory. Human health consequences are discussed in Section 4.3.6.6.4.

Table 4-45 Annual Incident-Free Airborne Radiological Emissions at the Idaho National Engineering Laboratory

Element	*Ci/yr*
Tritium + Carbon-14	3,100
Cesium-134 + Cesium-137	0.18
Cobalt-60	0.0000019
Plutonium	0.0077
Strontium-90 + Yttrium-90	0.058
Krypton-85	500,000
Antimony-125	16
Iodine-129 + Iodine-131	0.44
Others	0.21

Source: DOE, 1995b

4.3.6.6.3 Water Quality

Savannah River Site

DOE has analyzed the expected liquid radiological releases from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These releases are presented in Table 4-46 (Grainger, 1995). The health effects from these liquid releases are discussed in Section 4.3.6.6.4 below.

Idaho National Engineering Laboratory

Chemical separation activities at the Idaho National Engineering Laboratory would not affect water quality because the facility designs would prevent any accidental or incident-free discharge of liquid effluents (DOE, 1995c).

Table 4-46 Annual Incident-Free Liquid Radiological Releases at the Savannah River Site

Element	*Ci/yr*
Tritium	1.29
Strontium-89/90	0.013
Ruthenium-103/106	0.012
Cesium-137	0.033
Promethium-147	0.045

4.3.6.6.4 Occupational and Public Health and Safety

Potential exposures to workers and the public due to chemical separation activities were analyzed at both the Savannah River Site and the Idaho National Engineering Laboratory (DOE, 1995c). To estimate health effects, this analysis defined three receptor groups:

- onsite workers assigned to operations involving spent nuclear fuel,
- 1994 offsite population residing within an 80-km (50-mi) radius of the chemical separation facilities (exposure via air), and
- offsite population whom management site surface-water emissions could affect.

Each of these three receptor groups would receive an annual maximum individual dose and an annual population dose. The maximally exposed worker dose would be limited by regulation to 5,000 mrem per year, as in the basic implementation of Management Alternative 1.

Savannah River Site

Incident-Free Impacts at the Savannah River Site

The highest estimated incident-free dose rates for conventional chemical separation operations at the Savannah River Site are presented in Table 4-47 (DOE, 1995a). These chemical separation operations could include activities related to blending the separated HEU down to LEU and converting all LEU into an oxide suitable for long-term storage. Values in Table 4-47 represent the estimated dose rates due to these activities, including actual chemical separation, blending down, and conversion to oxide. Multiplying these values by the estimated program duration of 13 years yields the doses presented in Table 4-48. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-48. If the HEU were not blended down, but rather converted directly to oxide, the worker population dose would be higher because the conversion to oxide would take place in the Uranium Solidification Facility. In this facility, the workers would be closer to the uranium.

Table 4-47 Incident-Free Radiation Dose Rates Due to Chemical Separation at the Savannah River Site

	Maximum Individual Dose Rate (mrem/yr)	*Population Dose Rate (person-rem/yr)*
Public		
Via Air	0.66	27
Via Water	0.0098	0.033
Workers	5,000[a]	21

[a] *Assumed to be equal to the regulatory limit*

Table 4-48 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Savannah River Site

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	8.6	0.0000043	351	0.18
Via Water	0.13	6.4×10^{-8}	0.43	0.00021
Workers	65,000	0.026	273	0.11

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-8. The risks of storage at RBOF are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at RBOF for the full 13 years, the public MEI and population risks would be 7.1×10^{-10} LCF and 0.000036 LCF, respectively. These risks are much lower than the corresponding values in Table 4-48.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is equal to the 0.026 LCF in Table 4-48.

For the public, the estimated MEI risk from incident-free chemical separation activities would be 0.0000043 LCF. This risk means that an individual who lives at the Savannah River Site boundary would have an additional chance of less than one in one hundred thousand of incurring an LCF.

The handling-related worker population risk at RBOF is 0.10 LCF (see Table 4-14). This must be added to the 0.11 LCF from Table 4-48 to obtain the estimate of worker population risk due to chemical separation of foreign research reactor spent nuclear fuel. The estimated population risk for workers, including the handling-related risk, is 0.21 LCF, so there would be an approximately 21 percent chance of one additional LCF among the radiation workers.

The estimated total public population risk from chemical separation activities would be 0.18 LCF (see Table 4-48), which means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the Savannah River Site due to incident-free chemical separation activities.

Impacts of Chemical Separations Accidents at the Savannah River Site

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Savannah River Site (DOE, 1995a), including a hydrogen explosion in a high-level waste tank, an unpropagated fire in a solution vessel, two kinds of inadvertent transfers of solutions, a coil and tube failure in the cooling system, a nuclear criticality, a "red oil" explosion, a severe earthquake, and a tornado. The annual risks for the accident with the highest estimated combination of frequency and consequence are presented in Table 4-49. The most severe accident scenario is an unpropagated fire in a solution vessel. Multiplying these results by the estimated program duration of 13 years yields the risks presented in Table 4-50.

Table 4-49 Annual Impacts of Chemical Separation Accidents at the Savannah River Site

	Accident Frequency (per year)	Consequences (LCF)		Risks (LCF/yr)	
		Maximum Individual	Population	Maximum Individual	Population
Unpropagated Fire					
• Public	0.02	0.00018	1.3	0.0000036	0.026
• Workers	0.02	0.00086	---	0.000017	---

Table 4-50 Impacts of Accidents During Chemical Separation Operations at the Savannah River Site

	Maximum Individual Risk (LCF)	Population Risk (LCF)
Public	0.000047	0.34
Workers	0.00022	---

These results indicate that the estimated public MEI risk due to the chemical separation accidents is 0.000047 LCF. The estimated public population risk due to chemical separation accidents is 0.34 LCF. These risks must be combined with the risks of receiving/unloading and temporarily storing the foreign research reactor spent nuclear fuel, which were presented in Table 4-24. Assuming the foreign research reactor spent nuclear fuel would be received/unloaded and stored at RBOF for 13 years, the public MEI and population risks would be 0.0000026 LCF and 0.096 LCF, respectively.

The maximum of the two estimated accident-related MEI risks is 0.000047 LCF. This means that this hypothetical individual would have an additional chance of incurring an LCF of less than one in ten thousand.

The sum of the two population risks is 0.43 LCF. This means there would be an approximately 43 percent chance that one additional LCF would occur in the public population near the Savannah River Site due to accident conditions.

Idaho National Engineering Laboratory

Incident-Free Impacts at Idaho National Engineering Laboratory

The incident-free radiation dose rates for chemical separation at the Idaho National Engineering Laboratory are presented in Table 4-51 (DOE, 1995c). Multiplying these values by the estimated program duration of 12 years yields the doses presented in Table 4-52. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-52.

Table 4-51 Incident-Free Radiation Dose Rates due to Chemical Separation at the Idaho National Engineering Laboratory

	Maximum Individual Dose Rate (mrem/yr)	Population Dose Rate (person-rem/yr)
Public		
Via Air	0.048	0.39
Via Water	0.0	0.0
Workers	5,000[a]	18

[a] *Assumed to be equal to the regulatory limit*

Table 4-52 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	0.58	2.9×10^{-7}	4.7	0.0024
Via Water	0.0	0.0	0.0	0.0
Workers	60,000	0.024	216	0.086

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-9. The risks of storage are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at FAST for the full 13 years, the public MEI and population risks would be 2.5×10^{-9} LCF and 0.000021 LCF, respectively. These risks are much lower than the corresponding values in Table 4-52.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is higher than the 0.024 LCF in Table 4-52.

For the public, the estimated MEI risk due to incident-free chemical separation activities at the Idaho National Engineering Laboratory is less than one millionth of an LCF, which means that an individual who lives at the Idaho National Engineering Laboratory boundary would have an additional chance of less than one in a million of incurring an LCF.

The handling-related worker population risk at the Idaho National Engineering Laboratory is 0.10 LCF, from Table 4-15. This must be added to the 0.086 LCF from Table 4-52 to obtain the estimate of worker population risk due to incident-free chemical separation of foreign research reactor spent nuclear fuel, 0.19 LCF. This is near zero, so zero LCF would be expected among the radiation workers.

The estimated total public population risk at the Idaho National Engineering Laboratory is about 0.0024 LCF, which is much less than one LCF.

Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Idaho National Engineering Laboratory (DOE, 1995c). The accident with the highest estimated combination of frequency and consequence would be an inadvertent nuclear criticality during chemical separation. The accident frequency, consequences, and annual risks for this accident are presented in Table 4-53. Multiplying these results by the estimated program duration of 12 years yields the risks presented in Table 4-54.

Table 4-53 Annual Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

	Accident Frequency (per/yr)	*Consequences (LCF)*		*Risks (LCF/yr)*	
		Maximum Individual	*Population*	*Maximum Individual*	*Population*
Inadvertent Criticality					
Public	0.001	0.000025	0.0028	2.5×10^{-8}	0.0000028
Workers	0.001	0.0036	---	0.0000036	---

Table 4-54 Impacts of Accidents During Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Risk (LCF)	*Population Risk (LCF)*
Public	3.0×10^{-7}	0.000034
Workers	0.000044	---

The highest estimated public MEI risk is 3.0×10^{-7} LCF, which means that an individual living at the management site boundary would have an additional chance of incurring an LCF of less than one in a million.

The highest estimated public population risk is 0.000034 LCF, which is much less than one LCF.

4.3.6.6.5 Waste Management

Savannah River Site

DOE has analyzed the wastes that would be generated from the aluminum-based foreign research reactor spent nuclear fuel and from an additional inventory of aluminum-based spent nuclear fuel during chemical separation activities. High-level waste, saltstone, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Savannah River Site. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. These aluminum-based spent nuclear fuel elements are similar to DOE's Mark 16/22 spent nuclear fuel elements at the Savannah River Site.

High-level liquid waste would be transferred to the F/H-Area Tank Farm for volume reduction and then to the Defense Waste Processing Facility for conversion into a borosilicate glass form suitable for prolonged storage. The high-level glass waste that would result from chemically separating the 18.2 MTHM (approximately 17,800 elements) of foreign research reactor spent nuclear fuel in this implementation subalternative would fill about 72 canisters (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 200 canisters. These canisters would be managed with the estimated 5,717 canisters that the Savannah River Site expects to produce from the existing onsite inventory of liquid high-level waste (WSRC, 1995). Each canister will contain approximately 40,000 curies of radioactivity (DOE, 1994a). The representative radionuclide composition of the waste glass is presented in Table 2.11 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be cesium-137, strontium-90, and their daughters. DOE expects that this waste form would be acceptable for disposal in a geologic repository.

Saltstone would be produced during the vitrification of high-level waste at the Defense Waste Processing Facility. An estimated 4,000 m^3 (140,000 ft^3) would be generated from the 18.2 MTHM of foreign research reactor spent nuclear fuel and would be disposed of onsite (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 11,300 m^3 (400,000 ft^3). This is much less than the maximum estimated cumulative saltstone to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 625,211 m^3 (about 22,000,000 ft^3) (DOE, 1994b). The saltstone would contain far less radioactivity than the high-level waste glass: approximately 0.1 curie per cubic meter (DOE, 1994a). The approximate composition of the saltstone in terms of specific radionuclides is presented in Table C.5 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be promethium-147 until about 2000, then strontium-90 and its daughter thereafter.

Transuranic waste would not be generated during the chemical separation activities of foreign research reactor spent nuclear fuel (DOE, 1995a). The trace amounts of transuranic elements would not be removed from the waste stream, so they would be included in the high-level waste. If the Taiwan Research Reactor spent nuclear fuel (included in the total inventory of 51 MTHM) is chemically separated and the transuranic elements removed, then an estimated 832 m^3 (about 29,400 ft^3) of transuranic waste would be generated (DOE, 1995a). This is much less than the maximum estimated cumulative transuranic waste to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 9,426 m^3 (about 333,000 ft^3) (DOE, 1994b).

Hazardous/mixed waste would also be produced under this implementation subalternative. An estimated 104 m^3 (about 3,700 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a). This is much less than the maximum estimated cumulative mixed waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 14,720 m^3 (about 520,000 ft^3) (DOE, 1994b).

Solid low-level waste would also be produced under this implementation subalternative. An estimated 74,000 m^3 (about 2,600,000 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a) and would be disposed of onsite. This is much less than the maximum estimated cumulative low-level waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 397,177 m^3 (about 14,000,000 ft^3) (DOE, 1994b).

Idaho National Engineering Laboratory

DOE has also analyzed the wastes that would be generated from the foreign research reactor spent nuclear fuel and from an additional inventory of spent nuclear fuel during chemical separations activities at the Idaho National Engineering Laboratory. High-level waste, low-level grout, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Idaho National Engineering Laboratory. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 56 canisters of high-level waste glass would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel yields 90 canisters. Scaling up further to the total inventory of 65 MTHM yields an estimate of about 300 canisters. These canisters would be managed along with the estimated 8,500 canisters Idaho National Engineering Laboratory expects to produce from the existing inventory of high-level waste onsite (DOE 1995b). Although the waste form has not been determined yet, each canister is estimated to contain approximately 22,000 curies of radioactivity (DOE, 1994a). The composition of the waste form in terms of specific radionuclides has not been determined yet, but it is reasonable to expect it to be similar to that of the glass at the Savannah River Site. DOE expects that the waste form would be acceptable for disposal in a geologic repository.

Another possibility exists if large quantities of nonaluminum-based spent nuclear fuels are being chemically separated in these facilities. Some aluminum is necessary to produce the stable waste form, and the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel could satisfy this requirement. In this case, the chemical separation of the aluminum-based spent nuclear fuel would not increase the number of canisters that would be generated at the Idaho National Engineering Laboratory.

The estimates of low-level waste grout that would be generated under this implementation subalternative are also based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 1,280 m^3 (about 45,000 ft^3) of low-level waste grout would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel in this implementation subalternative yields about 2,000 m^3 (70,629 ft^3). Scaling up further to the total inventory of 65 MTHM yields an estimate of about 6,900 m^3 (about 245,000 ft^3). This grout would be managed along with the other grout the Idaho National Engineering Laboratory would produce onsite. The grout is expected to contain far less radioactivity than the high-level waste glass/ceramic: much less than one curie per cubic meter. The composition of the grout in terms of all the specific radionuclides has not been determined yet, but the major radioactive constituents would be cesium-137 and strontium-90. The cesium-137 and strontium-90 concentrations in the grout are expected to be about 0.034 and 0.0093 curies per cubic meter, respectively (Bendixsen, 1995).

Transuranic waste would not be generated during chemical separation of the foreign research reactor spent nuclear fuel. Furthermore, the Idaho National Engineering Laboratory would not separate the transuranic elements from the Taiwan Research Reactor spent nuclear fuel if it were transported there from the Savannah River Site. Therefore, no transuranic waste would be generated during chemical separation of the additional inventory of spent nuclear fuel. The estimated amount of cumulative transuranic waste for 10 years with minimum waste management at the Idaho National Engineering Laboratory is 67,000 m^3 (about 2,400,000 ft^3) (DOE, 1995c).

Hazardous/mixed waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 96 m^3 (3,400 ft^3) would be generated during 12 years of chemical separation operations. This is much less than the estimated 29,000 m^3 (1,020,000 ft^3) of cumulative hazardous and mixed waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c).

Solid low-level waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 68,300 m^3 (2,400,000 ft^3) would be generated during 12 years of chemical separation operations and would be disposed of onsite. This is more than the estimated 47,000 m^3 (1,660,000 ft^3) of low-level waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c). The Idaho National Engineering Laboratory would treat the waste at the Waste Experimental Reduction Facility and send it to the Radioactive Waste Management Complex for onsite disposal.

4.3.6.7 Summary of the Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-55 in terms of the risk of death due to cancer during each of the four segments of this implementation alternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The marine transport, port activities, and ground transport impacts are identical to the basic implementation of Management Alternative 1. The management site activity impacts were derived by comparing, and summing as appropriate, the handling impacts of the basic implementation of Management Alternative 1 and the impacts of chemical separation dedicated to foreign research reactor spent nuclear fuel. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-55 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The population risks were calculated by summing the appropriate spent nuclear fuel handling risks from the basic implementation of Management Alternative 1 with the risks of chemical separation at each management site and selecting the largest value. For example, the incident-free worker population risk of 0.21 LCF is the largest sum of the risks from that estimated for spent nuclear fuel handling operations under Phase 1 of the basic implementation of Management Alternative 1 and the estimated risk due to chemical separation dedicated to foreign research reactor spent nuclear fuel at the Savannah River Site or the Idaho National Engineering Laboratory. The sum of the above risks at the Idaho National Engineering Laboratory is 0.19 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-15) and 0.086 LCF from chemical separation], and the corresponding value at the Savannah River Site is 0.21 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-14) and 0.11 LCF from chemical separation].

As shown in Table 4-55, the total incident-free population risk would be 0.39 LCF for the potentially exposed public, while the corresponding risk would be 0.32 LCF for workers. Thus, there would be an estimated 39 percent chance of incurring 1 additional LCF among the exposed general public, and a 32 percent chance of incurring 1 additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-55. DOE and the Department of State estimate there could be about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

Table 4-55 Maximum Estimated Radiological Health Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.21	0.065
Accidents	1.3×10^{-11}	0.00014	---
Site Activities			
Incident-Free	0.026	0.18	0.21
Accidents	0.000047	0.43	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.39	0.32
Accidents	----	0.43	----

4.3.7 Implementation Alternative 7: New Developmental Treatment and/or Packaging Technologies

The environmental impacts of the developmental treatment and/or packaging technologies cannot be estimated with confidence at this time because the technologies and procedures are still under development. Implementation of certain of these technologies would require new facilities and thus would generate all the impacts associated with construction. Appropriate NEPA documentation would be prepared to support a decision on implementation of a new technology. The developmental treatment and/or packaging technologies are described in Chapter 2, Section 2.2.2.7.

The date at which a new facility would be operational is highly uncertain. A fairly simple technology implemented in existing facilities could be operational by 2000. On the other hand, the technology development, NEPA analysis, facility construction, and startup could take about 15 years for a complex technology. Thus, DOE could choose to implement one of the accept-and-store alternatives, in parallel with this alternative to prepare the foreign research reactor spent nuclear fuel for disposal. This may be necessary because foreign research reactor spent nuclear fuel may not be accepted in a geologic repository without some form of chemical processing or treatment. The repository acceptance criteria will not be final until a repository has been licensed.

Any new facilities would be designed to meet modern standards. The new design would minimize air and water emissions and the public and worker radiation doses at least as well as existing facilities, so DOE and the Department of State expect these impacts would be somewhat lower than those presented above for the conventional chemical separation technologies.

Some rough quantitative estimates are possible on the number of canisters that would be produced by some of the developmental technologies for disposal. Table 4-56 compares these estimates to the number of canisters that would be generated by chemical separation. The estimates of numbers of canisters that would be generated by the developmental treatment and/or packaging technologies do not depend on which DOE site performs the treatment and/or packaging.

Table 4-56 Comparison of Geologic Disposal Canisters for Various Technologies

Technology	*Approximate Number of Canisters*
Conventional Chemical Separation	
at the Savannah River Site	72
at the Idaho National Engineering Laboratory	90
Developmental Packaging Technologies	
Direct Disposal in Small Packages	140
Can-in-Canister	240
Developmental Treatment Technologies	
Melt and Poison	25
Chop and Poison	25
Melt and Dilute	180
Dissolve and Poison	950
Chop and Dilute	4,900
Dissolve and Dilute	11,800

The can-in-canister concept was recently introduced (Leventhal and Lyman, 1995), but it could be possible to implement it quickly at the Savannah River Site. Most of the foreign research reactor spent nuclear fuel elements would fit in cans of approximately 10 cm diameter and 85 cm length. If all of the approximately 22,700 elements were placed in these cans, the total canned volume would be about 150 m^3. Using the can-in-canister technology, this volume of glass would be displaced from high-level waste canisters to be produced in the Defense Waste Processing Facility. Since each canister has an internal volume of 0.625 m^3, displacing 150 m^3 of glass would require the production of approximately 240 additional high-level waste glass canisters at the Defense Waste Processing Facility.

The rest of the estimates of numbers of canisters that would be generated by the developmental technologies are scaled from a study (WSRC, 1994a) of the disposition of 7.3 MTHM of aluminum-based spent nuclear fuel, up to the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel. The melt and poison or chop and poison technologies could produce the fewest canisters, as low as 25 canisters. The consolidate and poison technology could produce the next lowest number of canisters (about 140) among the developmental technologies analyzed. The can-in-canister, melt and dilute, dissolve and poison, chop and dilute, and dissolve and dilute technologies would produce increasing numbers of canisters, in that order. The most canisters would be produced by the dissolve and dilute technology: over 11,000 canisters. This uncertainty in the number of canisters translates into a large uncertainty in the cost of disposal. Furthermore, it is not clear which, if any, of these waste forms would be acceptable in a geologic repository.

4.4 Management Alternative 2: Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

The basic implementation of Management Alternative 1 of the proposed action and the seven implementation alternatives to the basic implementation of Management Alternative 1 are all based on acceptance of foreign research reactor spent nuclear fuel into the United States. As discussed in Chapter 2, the two subalternatives under Management Alternative 2 facilitate overseas management of foreign research reactor spent nuclear fuel. This section discusses their policy considerations and environmental impacts. For convenience, the two subalternatives under Management Alternative 2 are defined briefly below:

1. Subalternative 1a - Overseas storage of the foreign research reactor spent nuclear fuel with U.S. technical and/or financial assistance, and

2. Subalternative 1b - Overseas reprocessing of the foreign research reactor spent nuclear fuel with U.S. nontechnical assistance.

Under these subalternatives, no foreign research reactor spent nuclear fuel would be accepted into the United States. The United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the spent nuclear fuel containing uranium enriched in the United States.

4.4.1 Subalternative 1a: Overseas Storage with U.S. Assistance

Policy Considerations

The foreign research reactor spent nuclear fuel could remain in interim storage overseas. The number of foreign research reactor spent nuclear fuel management sites involved would be greater and the quality of storage technology in some countries might be lower than if the basic implementation of Management Alternative 1, or one of its seven implementation alternatives, was adopted.

The cost of this subalternative might be greater than the cost of the basic implementation of Management Alternative 1 because it might not take advantage of economies of scale. To set up a secure area and a nuclear material handling infrastructure, purchase a storage cask, transfer the spent nuclear fuel to the cask, and maintain the secure area and nuclear infrastructure for 40 years would cost tens of millions of dollars. To repeat this in several dozen countries could potentially push the total cost up into the range of hundreds of millions of dollars. Furthermore, after incurring this expense, all of the U.S. origin HEU would still be located in foreign countries where a change in government could reverse any commitment to withhold the material from production of nuclear weapons.

This subalternative would be economically attractive only in countries that already have nuclear infrastructures. In these cases, the addition of the spent nuclear fuel from research reactors to existing spent nuclear fuel inventories in storage would involve only incremental costs without all the startup costs.

If the United States does not accept any near term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risks would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

For countries that will not allow the indefinite storage in their territories of increasing quantities of spent nuclear fuel, this subalternative is not a viable option. Under this scenario, reactor operators in these countries, in order to avoid shutting down, might be forced to consider storing their spent nuclear fuel in other countries, where safe and secure management and material accountancy problems could exist and the risk of illicit diversion could be a concern. For example, Austria was reportedly approached by commercial interests from Belarus with an offer to store spent nuclear fuel from the ASTRA reactor for hard currency. (Since the "Offsite Fuels Policy" for HEU spent nuclear fuel expired in 1988, the Austrian government has required that for fresh fuel to enter the country, an equivalent quantity of spent nuclear fuel must be shipped out of the country.) The offer, which was rejected in support of nuclear weapons nonproliferation policies, is indicative of the scenarios that may develop as pressure builds on reactor operators to close the back end of their nuclear fuel cycle.

Impacts

There would be no environmental impacts on U.S. territory for the duration of the interim period.

4.4.2 Subalternative 1b: Overseas Reprocessing with United States Non-Technical Assistance

4.4.2.1 Overview and Policy Considerations

Foreign research reactor spent nuclear fuel could be reprocessed in foreign facilities and the resulting high-level waste vitrified or cemented. No U.S. reprocessing technology would be used in this subalternative. The inventory and conditions for management of foreign research reactor spent nuclear fuel under Subalternative 1b are the same as those under basic implementation of Management Alternative 1. The amount of HEU that would be removed from international commerce is the same as under basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons)]. To be consistent with U.S. nuclear weapons nonproliferation policy, however, bilateral agreements would have to be established with one or more foreign governments before DOE and the Department of State could consider implementation of such a subalternative.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

There are four sites in Europe at which reprocessing is conducted for commercial customers: the Marcoule and La Hague sites in France, and the Dounreay and Sellafield sites in the United Kingdom. The companies that operate these sites are strictly regulated by their government agencies. The facilities at La Hague and Sellafield are dedicated to oxide spent nuclear fuel from commercial reactors and are not likely candidates for reprocessing the metallic foreign research reactor spent nuclear fuel. All four of these sites routinely release small quantities of radionuclides into the environment and produce radioactive wastes. For example, in 1993 the releases from the Dounreay facility to the North Sea included 2.7 Ci of total alpha activity, 220 Ci of beta activity excluding tritium, and 27 Ci of beta activity from tritium. These releases represented 13 percent, 7.2 percent, and 0.8 percent of the applicable regulatory limits (Jones et al., 1994). The radionuclides released into the atmosphere and into a river or sea would flow across international boundaries. These releases would cause a small, unmeasurable increase in world-wide natural background radiation levels. The transport of vitrified high-level waste away from the reprocessing facility would also produce environmental impacts on foreign territory and possibly in international waters.

Since the United States does not encourage the development of reprocessing capabilities overseas, DOE and the Department of State would only consider this subalternative in France or the United Kingdom where the capability already exists. Reprocessing would most likely take place (as it already has in several instances) at the Dounreay facility—the sole facility currently willing and able to reprocess foreign research reactor spent nuclear fuel. France's facility in Marcoule does reprocess spent nuclear fuel from French research reactors, but does not currently accept such spent nuclear fuel from other nations for reprocessing.

The British and French regulatory agencies require the customer to accept the wastes as a condition of reprocessing spent nuclear fuel, so this option would be unavailable to those countries lacking the technical or legal capability to store or dispose of high-level waste. Alternatively, the United States might consider accepting the wastes from reprocessing.

4.4.2.2 Waste Generation at the Foreign Reprocessing Site

Reprocessing the foreign research reactor spent nuclear fuel would produce two distinct streams: the uranium and the waste products.

For spent nuclear fuel containing HEU, the HEU would be blended down to LEU at the reprocessing facility. If the LEU were then shipped to the United States, the resulting environmental impacts would be no greater than for ordinary nonhazardous cargo because LEU produces such a small radiation dose rate.

The British and French have decades of experience in conditioning nuclear waste at their four reprocessing facilities. In recent years, they have greatly reduced the volumes of wastes that require disposal. Both nations use the same technology for vitrifying their high-level waste, and both nations produce the same size high-level waste glass canister: 0.15 m^3 (5.3 ft^3). These canisters of high-level waste glass are expected to be suitable for disposal in geologic repositories. As of September 1993, France and the United Kingdom had filled more than 2,100 and 350 canisters with high-level waste glass, respectively (Masson, et al., 1994).

As a general rule, European reprocessing and vitrification of about 8 to 10 MTHM of spent nuclear fuel would generate about 1 m^3 (35.3 ft^3) of high-level waste in glass form (UKAEA, 1994; Masson, et al., 1994). Thus, if all 19.2 MTHM of the foreign research reactor spent nuclear fuel were reprocessed and vitrified overseas, DOE and the Department of State estimate that the total volume of vitrified high-level waste would be only about 2.4 m^3 (85 ft^3). DOE and the Department of State estimate that the high-level waste from reprocessing all the foreign research reactor spent nuclear fuel would fill about 16 European-sized canisters. For reference, this volume of glass waste would fill four American-sized canisters.

4.4.2.3 Removal of Waste from the Reprocessing Site(s)

The British and French governments do not accept responsibility for ultimate disposal of the high-level waste glass canisters for foreign customers. Both nations require that disposal of the high-level waste glass canisters and any other wastes generated during reprocessing of their spent nuclear fuel, including low-level waste, be the responsibility of the nation(s) hosting the reactors. At the Dounreay Site, however, only small amounts of low-level waste have been generated during reprocessing of spent nuclear fuel from research reactors. Many nations with foreign research reactors, however, do not have any capabilities to accept the high-level waste glass canisters. The United States may accept the intact foreign research reactor spent nuclear fuel from these nations while simultaneously encouraging the nations which can accept the canisters to reprocess their foreign research reactor spent nuclear fuel under the conditions noted in Section 4.4.2.1. This would be a combination of the basic implementation of Management Alternative 1 and Subalternative 1b (overseas reprocessing) of Management Alternative 2.

As another option under this subalternative, if the host nations cannot accept this responsibility, the United States would commit to accept the high-level waste glass canisters. This could provide the incentive necessary to convince reactor operators to cooperate with the RERTR program and to use LEU in their reactors. Some nations may refuse to reprocess or require the United States to take title to the foreign research reactor spent nuclear fuel prior to reprocessing.

DOE and the Department of State could begin accepting canisters into the United States within the first 10 years, or DOE and the Department of State could specify that they be stored at the reprocessing facility for decades. If the canisters were accepted in the near term, they would most likely be stored at the Savannah River Site because this site has already built a new storage facility with a capacity of 2,286 canisters. If the canisters were stored overseas for decades, then they would be transported directly to the geologic repository.

Marcoule produces vitrified waste, similar to U.S. vitrified waste. In the United Kingdom on the other hand, as a result of a different regulatory structure, the wastes from reprocessing of research reactor spent nuclear fuel are classified as intermediate-level radioactive wastes. (In the United States, these same materials would be classified as high-level radioactive wastes.) In the United Kingdom, the intermediate-level wastes are mixed with a special cement and poured into steel drums, which can then be buried. This waste form is dissimilar to the vitrified borosilicate glass high-level waste form that is expected to be produced in the United States, and is incompatible with United States radioactive waste disposal standards. The government of the United Kingdom might allow an exchange of vitrified commercial waste from Sellafield for cemented waste from Dounreay, which might allow the United States to accept vitrified high-level waste from the United Kingdom.

Transportation of vitrified high-level waste must conform to U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR 71) regulations. Under this option, the European-sized glass canisters would be transported in "Type B" casks, which provide a high degree of assurance that cask integrity will be maintained with essentially no loss of radioactive contents or serious impairment of the shielding capability provided by the cask, even in severe accidents. DOE has prepared initial designs for a defense high-level waste cask for truck transportation of the Savannah River Site high-level waste. As initially designed, the defense high-level waste cask uses a solid body concept to absorb energy during an accident and normal transportation conditions. To minimize the exposure to gamma radiation, shielding would be provided by a depleted uranium liner inside the cask body. (Gamma radiation is high-energy, short wavelength electromagnetic radiation with properties similar to x-rays.) The regulatory limit for radiation dose rate outside the cask is 10 mrem per hour at 2 m (6.6 ft) from the edge of the vehicle. Casks transported under this option are assumed to emit this level of radiation. Currently, however, no casks for shipping high-level waste canisters by truck or rail have been certified by the NRC.

Each of these "Type B" casks would be large enough to hold two European-sized glass canisters. Thus, the option of overseas reprocessing with acceptance of approximately 16 high-level waste glass canisters would require about 8 cask shipments into the United States (versus 721 cask shipments by sea and 116 by land under the basic implementation of Management Alternative 1). Vitrified high-level waste shipments would use the same East Coast port(s) identified in Chapter 2 for foreign research reactor spent nuclear fuel. The same procedures and representative overland routes analyzed for foreign research reactor spent nuclear fuel would apply to these shipments of vitrified high-level waste. The management site for these canisters would be the Savannah River Site. Alternatively, they might be transported directly to the candidate geologic repository at Yucca Mountain, NV.

Each of the eight casks is assumed to contain the waste products associated with one-eighth of the foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1.

Marine Transport Impacts

Risks under Subalternative 1b were assessed using the same methodology used to evaluate risks associated with the transport of the foreign research reactor spent nuclear fuel. The major differences in the analysis are the number of cask shipments and the isotopic content within each transportation cask.

Impacts of Incident-Free Marine Transport

As with the shipment of foreign research reactor spent nuclear fuel, the primary impact of incident-free marine shipping of vitrified waste would be upon the crews of the ships. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste. The primary contribution to the crew dose would come from the daily cargo inspection activities. Three crew members have been modeled as performing the inspections and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis it has been assumed that the vitrified waste would be transported on a chartered vessel, there would be no intermediate port calls, and the shipment would originate in Europe (either the United Kingdom or France.)

As in the spent nuclear fuel analysis, either two or eight casks are assumed to be on each single voyage. This assumption results in exposure to two radiation fields during all activities that bring crew members into the vicinity of the transportation casks. Should all the casks be shipped at once, this assumption is equivalent to assuming that this single voyage is made with two casks per hold in one vessel. The crew risk would be the same for this single voyage as for four voyages with two casks per vessel.

Results of the marine incident-free risk analysis are presented in Table 4-57. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the crew would be approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, would be well below the DOE and NRC limits for public exposure of 100 mrem per year. If, however, all the casks were shipped in 1 year (perhaps all on one ship), then the maximally exposed worker dose would exceed the limit of 100 mrem per year. In this case, new inspectors would be used to keep each individual's dose below the limit.

Table 4-57 Incident-Free Marine Transport Impacts (Subalternative 1b)

	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Crew Risk (LCF)*
Per voyage (2 casks)	53	0.000021	0.19	0.00007
Entire program	210	0.000084	0.74	0.00030

Impacts of Accidents During Marine Transport

If the ship carrying a cask of vitrified waste were to catch fire at sea and the cask was sufficiently damaged by fire to release its contents, members of the ship's crew near the fire would be exposed to the released radioactive material. Any resulting plume carrying radioactive particles would disperse over the ocean, where there is no human population. Therefore, the ship's crew would be the only people exposed to the released radioactive material. The number of ship's crew members is considerably smaller than the population modeled within a short distance of an accident that occurs in the port. Therefore, consequences of a shipboard accident resulting in the release of radioactive material in a plume would be covered by the consequences of the accidents considered in the port analysis. As discussed below, because the oceans are a very dilute system, effects on marine biota would not be discernible.

If a collision or other accident (e.g., loss of a cask over the side in a storm) occurred in which an intact cask fell overboard, the fact that the cask would be immersed would not necessarily result in a release of its contents. Spent nuclear fuel casks are designed to withstand at least a 15-m (50-ft) immersion, and it has been demonstrated that the cask seals will remain intact at much greater depths (IAEA, 1990). Spent nuclear fuel casks, damaged or undamaged, can be recovered from water up to 200 m (660 ft) deep: well beyond the range typical of coastal and port depths. (Recovery at great depths, e.g., more than 2,000 m or 6,600 ft, is possible, but would be costly). It is reasonable to believe that a cask would be recovered in any incident involving the immersion of a cask in waters up to 200 m (660 ft) in depth.

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development, Paris, France, estimated the impacts of various accident scenarios involving shipment of reprocessed commercial spent nuclear fuel. The Nuclear Energy Agency estimated that a damaged and unrecovered cask of high-level waste in coastal waters would result in a peak individual human dose of 6.5 mrem per year per MTHM (NEA, 1988). Dose and exposure estimates that follow are based on the estimates generated in the Nuclear Energy Agency study and are modified to take into account the content of the casks based on the shipment of all material from the foreign research reactor spent nuclear fuel program in eight cask shipments.

In the most extreme situation, where the accident occurs in coastal waters, the spent nuclear fuel is not recovered, and the cask is damaged, the peak dose to an individual human is estimated to be 19 mrem per year. The individual is assumed to reside near the shore and ingest seafood (fish, mollusk, and seaweed) harvested from the area in the immediate vicinity of the vitrified waste transportation cask. (For an initially intact cask, the dose would be expected to be considerably lower, approximately 0.3 mrem per year.) Peak biota doses are estimated at 0.8 mrad per year for fish, 0.9 mrad per year for crustaceans, and 19 mrad per year for mollusks, if the cask were damaged and not retrieved from coastal waters. With cask retrieval, both the peak dose to an individual and the biotic impacts would be considerably smaller. The results for the loss and failure of a single cask are lower than the peak impacts for the loss and failure of a single spent nuclear fuel cask (see Section 4.2.1.3), principally due to the lower leach rate for vitrified waste (see Appendix C).

In deep waters, the radioactive constituents of the vitrified waste would be released slowly over time into the surrounding waters if the cask were not recovered. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a damaged cask of vitrified high-level waste were submerged on the deep ocean bottom, the peak human individual dose to an individual residing along the coast and ingesting seafood harvested from the general area in which the breached submerged cask is located is estimated to be 0.000015 mrem per year.

Humans would not be the principally exposed species in a deep ocean accident involving vitrified waste casks. Using the Nuclear Energy Agency estimates and assuming that the damaged waste cask lay on the ocean floor where it slowly released its radioactive inventory, the peak doses to biota residing on the ocean floor in or near the uppermost sediment layer would be 0.9 rad per year for fish, 1.2 rad per year for crustaceans, and 41 rad per year for mollusks (NEA, 1988).

Harmful effects of chronic irradiation have not been observed in natural aquatic populations at dose rates less than 365 rad per year (NCRP, 1991). At doses an order of magnitude below this, as would be the case in an accident involving the vitrified waste from the foreign research reactor spent nuclear fuel, it is unlikely that either a population of marine biota or individual members of that population would be harmed by the radiation resulting from a spent nuclear fuel accident. Furthermore, no chemical hazard would be expected from the release of the contents of the vitrified waste canisters into the open ocean.

Using the same accident probabilities used in the marine transport analysis of the basic implementation of Management Alternative 1, risk estimates were developed for this subalternative. The MEI risk due to the loss of a vitrified high-level waste cask in the ocean is very low for the shipment of up to eight casks. The highest estimated risk to a human would occur in the accident scenario in which a cask is sunk and not recovered from coastal waters. This scenario would result in an MEI risk on the order of 1×10^{-10} mrem per year, which corresponds to about 2.7×10^{-15} LCF. This means that the chance of the MEI incurring one LCF due to this subalternative would be about one in one quadrillion.

Port Activity Impacts

Impacts of Incident-Free Port Activities

As with the shipment of the foreign research reactor spent nuclear fuel, the primary impact of incident-free port activities required to unload the vitrified waste casks is upon the workers: port handlers, inspectors, and port staging personnel. Most of the assumptions used in the analysis of the port worker exposure to the foreign research reactor spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste.

Results of the port activities' incident-free risk analysis are presented in Table 4-58. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the port workers are approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, are well below the DOE and NRC limits for public exposure of 100 mrem per year.

Impacts of Accidents During Port Activities

The methodology used to evaluate the accident consequences and risks associated with port accidents is identical to that used to assess these items for the shipment of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1 (Section 4.2.2.3). The MACCS code was used with site-specific population and meteorology data to determine the consequences of an accident. The inventory (radionuclide content) of the transportation casks was determined by combining the radionuclide content of all of the vitrified waste to be returned to the United States under this subalternative and equally dividing it among the eight casks. In this analysis it was assumed that the Canadian spent nuclear fuel, which was assumed to be sent to the United States via truck in the analysis documented in Section 4.2.2.3, would be sent to Europe and reprocessed. The vitrified waste from this spent nuclear fuel is included in this analysis.

Table 4-58 Incident-Free Port Activity Impacts (Subalternative 1b)

Impacts Per Shipment				
	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed WorkerRisk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Inspectors	1.3	5.2×10^{-7}	0.0053	0.0000021
Port Handlers	0.46	1.8×10^{-7}	0.0015	0.00000061
Port Staging Personnel	0.38	1.5×10^{-7}	0.0046	0.0000018
Maximum	**1.3**	$\mathbf{5.2 \times 10^{-7}}$	----	----
Total	----	----	**0.011**	**0.0000042**
Impacts for the Entire Subalternative 1b				
	Maximally Exposed WorkerDose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Workers (person-rem)*	*Population Risk to Workers (LCF)*
Inspectors	10	4.0×10^{-6}	0.04	0.000017
Port Handlers	4	1.6×10^{-6}	0.01	0.0000048
Port Staging Personnel	3	1.2×10^{-6}	0.04	0.000015
Maximum	**10**	$\mathbf{4.0 \times 10^{-6}}$	----	----
Total	----	----	**0.09**	**0.000036**

The amounts of material released from the glass in various accident scenarios, called release fractions, are based on information developed for accident analysis at the Savannah River Site (DOE, 1994k). These release fractions are the same for the three accident categories analyzed for the spent nuclear fuel port accidents (the accident categories included collisions and collisions followed by fires). Therefore, these three accident categories were combined to form a single category for this analysis. Accident probabilities were developed for this single accident category at both the dock and in the approach to the dock.

Since all of the vitrified waste would be transported to the United States from Europe, only East Coast ports were selected for port-specific analysis of the accident consequences. The port accident analysis was performed for three East Coast ports: Philadelphia, PA; Charleston, SC; and MOTSU in North Carolina. These three ports represent a wide range of port city populations. As in the port accident analysis discussed in Section 4.2.2.3, these ports are not necessarily the selected ports of entry for the vitrified waste. They are intended to be representative of the range of populations, and therefore consequences, associated with all of the potential ports of entry.

Results of this analysis are presented in Table 4-59. The consequences of an accident in port involving a cask of vitrified waste would be lower than for the category 5 and 6 accidents involving the foreign research reactor spent nuclear fuel casks. This is a result of the much lower release fractions associated with the vitrified waste compared to the release fractions for the metallic spent nuclear fuel. However, for the vitrified waste, category 4 accidents result in the release of the same amount of material from the vitrified waste as the category 5 and 6 accidents. (For foreign research reactor spent nuclear fuel, category 4 accidents result in much smaller consequences.) Because the frequency of this category of accidents is two orders of magnitude higher than that for category 5 and 6 accidents, the port accident risks per single-cask shipment are higher for vitrified waste than for foreign research reactor spent nuclear fuel. The port accident population risks would be about the same order of magnitude as those under the basic implementation of Management Alternative 1 because of these category 4 accidents.

Table 4-59 Port Accident Risks (Subalternative 1b)

Port	Risk per Single-Cask Shipment of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

The MEI doses calculated for these accidents have a rather small variance. The largest estimated MEI dose is 740 mrem. The largest probability of one LCF (given that the accident has occurred) was 0.00035. Combining these estimates with the probability of a severity category 4 accident per shipment and the number of shipments results in an MEI risk of 1.8×10^{-8} LCF.

Ground Transport Impacts

Under Subalternative 1b, DOE and the Department of State would transport eight casks of vitrified high-level waste overland from an East Coast port(s) to a candidate geologic repository (in Nevada for example). The shipments may go directly from the port(s) to the candidate geologic repository or they might go from the ports to the Savannah River Site for storage, then from the Savannah River Site to the candidate geologic repository. Results are displayed in Figures 4-18 and 4-19.

Impacts of Incident-Free Ground Transport (Ports to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate near vehicles carrying vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. Incident-free transportation of vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00023 to 0.0032 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.0001 to 0.0008. The estimated number of radiation-related LCF for the general population ranged from 0.00009 to 0.0024, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005. The impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

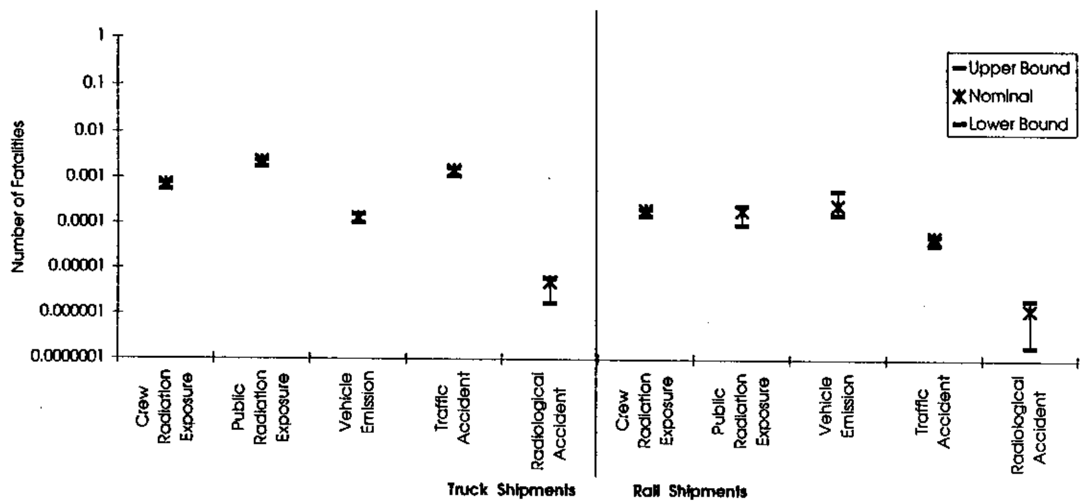
To estimate the maximally exposed ground transport worker risk, DOE and the Department of State assumed all the vitrified waste was transported during a 1-year period and one truck driver received his annual limit of 100 mrem during that year. This dose translates into a risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.0000002 to 0.0000059 LCF from radiation and from 0.00003 to 0.0016 for traffic fatality, depending on the transportation mode and the port(s) selected.

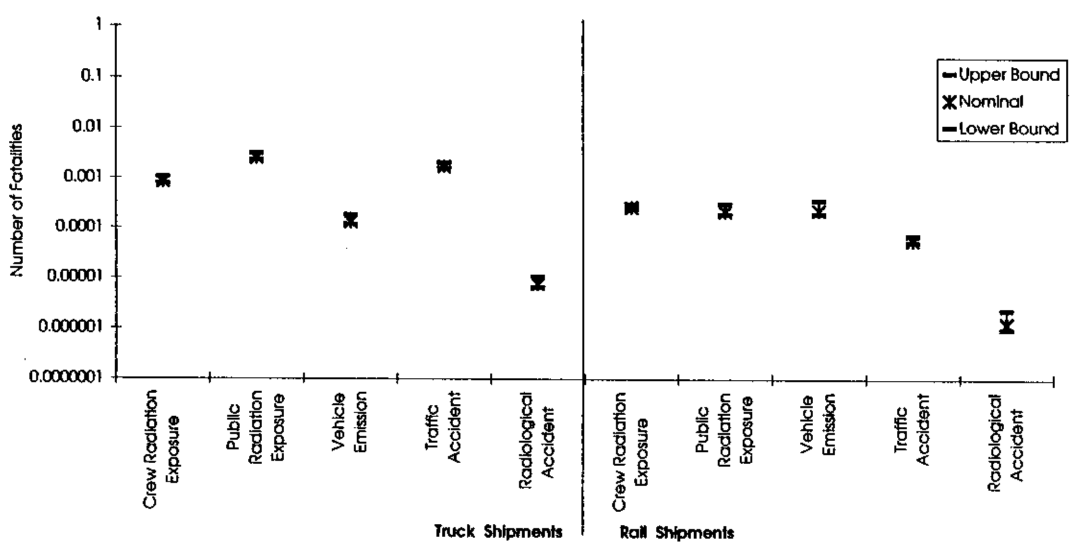
Impacts of Incident-Free Ground Transport (Ports to the Savannah River Site to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate from casks containing vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. The



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-18 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Repository)



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-19 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Savannah River Site to Repository)

incident-free transportation of the vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00041 to 0.004 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00023 to 0.001. The estimated number of radiation-related LCF for the general population ranged from 0.00018 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.00011 to 0.00035. Impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed worker risk, it was assumed that the two legs of ground transport would be separated by a long storage period. That is, the second leg (transport from the Savannah River Site to the repository) would occur at least 20 years after the first leg (transport from the ports to the Savannah River Site). Thus, one individual truck driver would probably not be involved in both legs. DOE and the Department of State further assumed that each leg would last no more than 1 year, so no individual truck driver could receive more than the annual regulatory limit of 100 mrem. This translates into a maximally exposed worker risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to the Savannah River Site to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.000001 to 0.00001 LCF from radiation and from 0.00005 to 0.002 for traffic fatality, depending on the transportation mode and the port(s) selected.

The consequences of the maximum foreseeable offsite transportation accident are greater than those of the basic implementation of Management Alternative 1. The frequency, however, is lower due to the reduced amount of ground transport. Maximum estimated MEI risk is reduced to 7×10^{-12} LCF.

Management Site Impacts

Impacts of Incident-Free Management Site Activities

Environmental impacts associated with the receipt and storage of the vitrified high-level waste canisters under Subalternative 1b are limited to the exposure of the working crew that would handle the incoming canisters at the site. The 16 canisters of vitrified waste (approximately 0.15 m^3 or 5.3 ft^3 each) would be received in 8 shipping casks and stored at the Glass Waste Storage Building at the Savannah River Site. The facility, described in Appendix F, has been designed for vitrified waste and has space for 2,286 canisters. Vitrification of all existing liquid high-level waste at the Savannah River Site is expected to produce a total of approximately 5,717 canisters. The impact of this additional amount of glass waste on the operational characteristics of the facility would be very low.

Vitrified waste would not contain any gaseous fission products, so there is no mechanism for incident-free emissions of radioactive material. Thus, impacts to the public near the Savannah River Site under this subalternative would be equal to zero.

To estimate the maximally exposed worker dose, DOE and the Department of State assumed that all the canisters would be received during one year. This is reasonable because of the small number of cask shipments. Then DOE and the Department of State conservatively assumed that one of the workers involved in handling these shipments would receive the maximum annual dose of 5,000 mrem allowed by regulation. This dose translates into an increased risk of 0.002 LCF.

The population dose to workers handling the eight casks would be 2.6 person-rem, based on the methodology presented in Appendix F, Section F.5 for unloading and storing in a vault-type dry storage structure. This translates into a worker population risk of 0.001 LCF.

Impacts of Accidents Onsite

The addition of 16 European-sized canisters to the thousands of larger American-sized canisters is expected to increase the accident risk by a very small increment, so this increase in the risk was not specifically analyzed in this EIS. The accident analysis for the Defense Waste Processing Facility has been reported in its Final EIS (DOE, 1994e).

Since vitrified waste contains no gaseous fission products, however, it is clear that the spent nuclear fuel element breach accident scenarios are not applicable to this subalternative. Thus, the aircraft-crash-with-fire scenario would present the highest risks. The highest annual estimates of MEI/NPAI and population risks under the basic implementation of Management Alternative 1 for this accident scenario are 1.2×10^{-9} LCF and 0.0000015 LCF, respectively (see Section 4.2.4.1). DOE and the Department of State consider these estimates to cover the risks for vitrified waste because the vitrified waste is designed to be much more stable than spent nuclear fuel in all accidents. Multiplying these annual estimates by the number of years the accident might occur (30 years) yields the risks for this alternative: 3.6×10^{-8} LCF for the MEI/NPAI risk and 0.000045 LCF for the population risk.

4.4.2.4 Disposal Site Impacts

Whether the vitrified high-level waste canisters were managed at the Savannah River Site or in Europe, eventually they would be transported to a geologic repository for disposal under this subalternative. Current planning for the U.S. candidate geologic repository at Yucca Mountain in Nevada indicates that acceptance of high-level waste canisters would begin early enough that the high-level waste from foreign research reactor spent nuclear fuel could be shipped to and emplaced in the repository before the end of the interim period.

Impacts due to handling European-sized canisters at the repository would be similar to the impacts due to handling American-sized canisters. After emplacement in the disposal site, no more impacts are expected to workers, the public, or the environment for at least 10,000 years because the radioactive material would be extremely unlikely to escape from the repository.

4.4.2.5 Summary of the Impacts of Subalternative 1b

The principal impacts under Subalternative 1b would be occupational and public health and safety impacts. These impacts would be due to the acceptance of vitrified high-level waste into the United States from Europe. (If no high-level waste were accepted, then there would be no impacts on U.S. territory.) These impacts are presented in Table 4-60 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-60 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of high-level waste producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing

Table 4-60 Maximum Estimated Radiological Health Impacts of Subalternative 1b

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.000084	0	0.0003
Accidents	2.7×10^{-15}	much less than 0.00002	---
Port Activities			
Incident-Free	0.000004	0	0.000036
Accidents	1.8×10^{-8}	0.00002	---
Ground Transport			
Incident-Free	0.00005	0.003	0.001
Accidents	7×10^{-12}	0.00001	---
Site Activities			
Incident-Free	0.002	0	0.001
Accidents	3.6×10^{-8}	0.000045	---
Highest Individual Risk			
Incident-Free	0.002	----	----
Accidents	3.6×10^{-8}	----	----
Total Population Risk			
Incident-Free	----	0.003	0.0027
Accidents	----	0.000075	----

people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) during the 1 year of high-level waste acceptance.

The highest estimated incident-free individual risk is 0.002 LCF, which would apply to an onsite radiation worker. This individual would have a one in five hundred chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally worker risk. DOE estimates this risk to be very nearly zero LCF.

The maximum estimated accident MEI risk is 3.6×10^{-8} LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten million. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The total incident-free population risk for both the general public and workers would be much less than one LCF.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-60. There is about a 0.2 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.5 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

As discussed in Section 2.4, DOE and the Department of State could combine implementation elements from Management Alternatives 1 and 2. Analysis of this example Hybrid Alternative does not signify its preference over other possible Hybrid Alternatives.

Under this Hybrid Alternative, DOE and the Department of State would facilitate reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2. It is assumed that the foreign research reactor operators in countries that can accept the reprocessing waste would agree to this arrangement. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States as in Management Alternative 1. (Refer to Section 2.4 for a more detailed description of this Hybrid Alternative).

Based on the current capabilities of overseas reprocessors, and for purposes of this analysis, only aluminum-based foreign research reactor spent nuclear fuel is assumed to be considered for reprocessing; all TRIGA spent nuclear fuel is assumed to be stored in the United States.

Under the Hybrid Alternative, the aluminum-based foreign research reactor spent nuclear fuel to be managed in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States), discussed in Sections 2.2.2.6 and 4.3.6. The uranium and waste products from this chemical separation would be managed as described in Sections 2.2.2.6 and 4.3.6, and the impacts of these activities would be covered by the impacts presented in those sections. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory where it would be stored at existing storage facilities until ultimate disposition. This distribution of the spent nuclear fuel is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

The environmental impacts associated with the foreign research reactor spent nuclear fuel that would be accepted into the United States, and the policy considerations of the Hybrid Alternative, are discussed below.

Policy Considerations

Under the Hybrid Alternative, up to 5.3 MTHM and about 5,600 elements of foreign research reactor spent nuclear fuel would be reprocessed overseas. The rest of the foreign research reactor spent nuclear fuel included in the basic implementation of Management Alternative 1, up to 13.9 MTHM and about 17,100 elements, would be accepted into the United States. Overall, the same amount of HEU as in the basic implementation of Management Alternative 1 would be removed from international commerce, up to about 4.6 metric tons (5.1 tons) of HEU.

4.5.1 Marine Transport Impacts

Impacts of Incident-Free Marine Transport

Impacts of incident-free marine transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. Incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.021 to 0.024 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed under the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The highest estimate of the incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF for all the shipments combined).

Impacts of Accidents During Marine Transport

Population risks due to accidents under the Hybrid Alternative would be reduced from those associated with the basic implementation of Management Alternative 1 because of the reduced amount of marine transport. As before, the population risks of accidents at sea are bounded by the risk of accidents in port.

The maximum consequences of the at-sea accidents for the Hybrid Alternative are no different than those of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments compared to the basic implementation of Management Alternative 1, the likelihood of such an accident would also be reduced. The Hybrid Alternative would require approximately 63 percent of the number of shipments required under the basic implementation of Management Alternative 1. The highest estimated risk due to an accident during marine transport would therefore be 0.00012 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of about 3×10^{-10} LCF. This means that this individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

4.5.2 Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. The Hybrid Alternative would require about 63 percent of the number of cask shipments required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities are proportionally reduced. The estimated number of LCF associated with this alternative range from 0.0021 to 0.0076. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The highest estimate of incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF).

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. Port accident risks associated with the Hybrid Alternative are estimated to range from 2×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

Consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of cask shipments, so the MEI risk is reduced to about 1 x 10^{-10} LCF.

4.5.3 Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The results are presented in Figure 4-20. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.011 to 0.15 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and the possibility of using different ports that created varying shipment distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.008 to 0.037. The estimated number of radiation-related LCF for the general population ranged from 0.010 to 0.11, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0031 to 0.025. Since these risk numbers are much less than one, implementation of the Hybrid Alternative would be unlikely to result in one LCF.

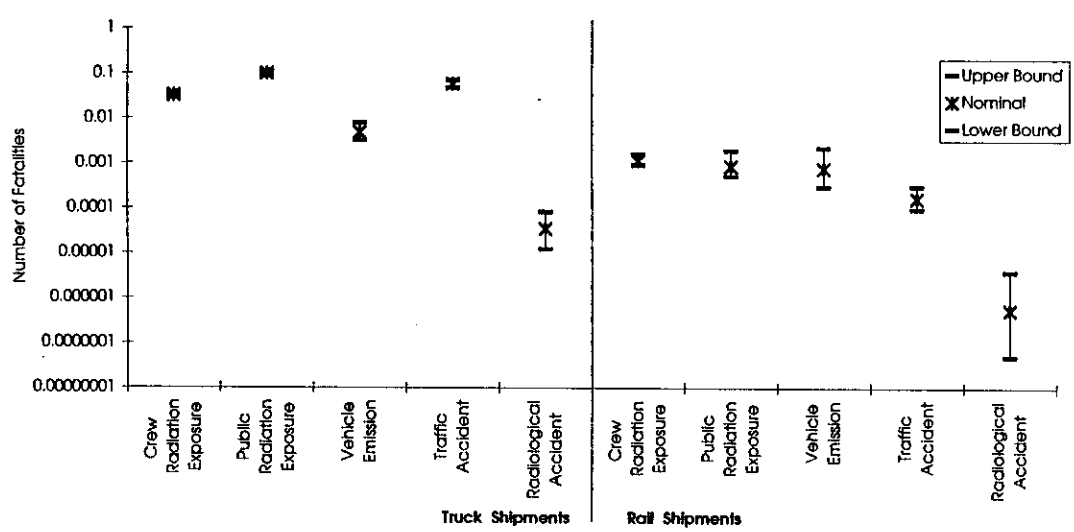


Figure 4-20 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 3 (the Hybrid Alternative)

Impacts of Accidents During Ground Transport

Transportation accident population risks over the entire Hybrid Alternative are estimated to range from 0.000005 to 0.000081 LCF from radiation and from 0.002 to 0.069 for traffic fatality, depending on the transportation mode and the ports that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 7.1×10^{-12} LCF due to the reduced amount of ground transport.

4.5.4 Management Site Impacts

Under the Hybrid Alternative, the amount of foreign research reactor spent nuclear fuel that would be accepted into the United States is about 17,100 elements and 13.9 MTHM. All the TRIGA spent nuclear fuel, representing approximately 4,900 elements and 1.0 MTHM, would be received and stored in existing facilities at the Idaho National Engineering Laboratory. Aluminum-based spent nuclear fuel, representing approximately 12,200 elements and 12.9 MTHM, would be received and chemically separated at the Savannah River Site as described in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States). Environmental impacts associated with the receipt and storage of the TRIGA spent nuclear fuel at existing facilities at the Idaho National Engineering Laboratory would be covered by the impacts presented for the basic implementation of Management Alternative 1 without construction of new facilities (Section 4.2). Environmental impacts associated with the receipt and chemical separation of the aluminum-based spent nuclear fuel at the Savannah River Site would be covered by the impacts presented for the near-term chemical separation alternative at the Savannah River Site (Section 4.3.6). The occupational and public health and safety impacts for both sites were estimated by combining the appropriate results from earlier analyses for the Idaho National Engineering Laboratory and the Savannah River Site.

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 elements that would be received and managed at the Idaho National Engineering Laboratory under this alternative represent about 22 percent of the number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the Hybrid Alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Multiplying the results in Table 4-9 by the maximum duration of each activity (13 years for receipt and 40 years for storage) yields the highest estimated risks for this part of the Hybrid Alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

The approximately 12,200 elements that would be received at the Savannah River Site under this alternative represent about 54 percent of the number of elements that would be received and temporarily stored there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions during receipt at the Savannah River Site under the basic implementation of Management Alternative 1 are presented in Table 4-8. The impacts for storage in RBOF are much smaller than those for receipt. Multiplying these results by 54 percent and the maximum duration of 13 years yields the highest estimated risks for this part of the Hybrid Alternative. The highest estimated public MEI risk is 3.9×10^{-10} LCF and the corresponding estimated public population risk is 0.000020 LCF.

The approximately 12.9 MTHM that would be chemically separated at the Savannah River Site under this alternative represents about 71 percent of the MTHM that would be chemically separated there under Implementation Alternative 6 dedicated to foreign research reactor spent nuclear fuel. Public impacts due to this implementation alternative were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated impacts to the public near the Savannah River Site due to this part of the Hybrid Alternative. Using this procedure, the highest estimated incident-free public MEI risk at the Savannah River Site is 0.0000031 LCF. The highest estimated incident-free public population risk at the Savannah River Site (including both the air and water exposure pathways) is 0.13 LCF.

The maximum of the three onsite activities' estimated public incident-free MEI risks is equal to 0.0000031 LCF, which would result from chemical separation activities at the Savannah River Site (The three parts are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to the Hybrid Alternative would be less than one in one hundred thousand.

The total of the three onsite activities' estimated public incident-free population risks is 0.13 LCF.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of this Hybrid Alternative is 13 years, the same as that in both the basic implementation of Management Alternative 1 and Implementation Alternative 6. Thus, the estimated maximally exposed worker dose is also the same. The maximally exposed worker risk is estimated to be 0.026 LCF.

Incident-free worker population impacts due to the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4. Using the same evaluation process described in Appendix F, Section F.5, for the 162 casks of TRIGA foreign research reactor spent nuclear fuel that would be received and unloaded under this Hybrid Alternative yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the Hybrid Alternative is 0.021 LCF.

Workers at the Savannah River Site would receive and unload 406 casks of aluminum-based foreign research reactor spent nuclear fuel in an existing wet facility under this alternative, receiving a population dose of 157 person-rem. The associated worker population risk for this part of the Hybrid Alternative is 0.063 LCF.

Incident-free worker population impacts due to Implementation Alternative 6 (chemical separation) were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated incident-free impacts to the workers at the Savannah River Site due to the Hybrid Alternative. Using this procedure, the highest estimated incident-free worker population risk due to chemically separating this spent nuclear fuel at the Savannah River Site is 0.078 LCF.

The total of the three onsite activities' estimated incident-free worker population risks is 0.16 LCF.

Impacts of Accidents Onsite

Accident scenarios, frequencies, consequences, and annual risks for the Hybrid Alternative are derived from those for the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory and Implementation Alternative 6 at the Savannah River Site.

Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory were presented earlier in this chapter in Table 4-25. Multiplying these by the duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory under this alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the mixture of TRIGA and aluminum-based spent nuclear fuel in the basic implementation of Management Alternative 1. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 1.9×10^{-6} LCF, which is due to an accidental criticality in a wet storage facility. The highest estimated accident population risk for this part of the Hybrid Alternative is 0.0088 LCF, which is due to the same accident scenario.

Annual accident risks for receipt and unloading at the Savannah River Site were presented in Table 4-24. Multiplying these by the duration of the receipt activity (13 years) yields the risk due to receipt and temporary storage accidents under this alternative. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 2.6×10^{-6} LCF, which is due to an accidental criticality in RBOF. The corresponding estimated accident population risk for this part of the Hybrid Alternative is 0.096 LCF.

The accident MEI and population impacts due to chemical separation were presented earlier in this chapter in Table 4-50. Multiplying these results by 71 percent yields the estimated impacts at the Savannah River Site due to accidents under the Hybrid Alternative. Using this procedure, the highest estimated public MEI risk due to accidents during chemical separation at the Savannah River Site is 0.000033 LCF. The highest estimated accident population risk at the Savannah River Site is 0.24 LCF.

The maximum of the three onsite activities' estimated accident public MEI risks is equal to 0.000033 LCF, which would occur at the Savannah River Site. (The three onsite activities are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to this Hybrid Alternative would be less than one in ten thousand.

The total of the three onsite activities' estimated accident population risks is 0.34 LCF.

4.5.5 Summary of the Impacts of the Hybrid Alternative

Principal impacts of the Hybrid Alternative would be occupational and public health and safety impacts. These are presented in Table 4-61 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population. In general, however, the implementation of the Hybrid Alternative would not pose higher risks than those determined for Management Alternative 1, assuming identical United States site management technology implementation. This is because the analyses in Management Alternative 1 aud its implementation alternatives considered the management of the maximum amount of foreign research reactor spent fuel in the United States.

Table 4-61 Maximum Estimated Radiological Health Impacts of the Hybrid Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.024
Accidents	3×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00052	0	0.0076
Accidents	1×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00052	0.11	0.037
Accidents	7.1×10^{-12}	0.000081	---
Site Activities			
Incident-Free	0.026	0.13	0.16
Accidents	0.000033	0.34	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.000033	----	----
Total Population Risk			
Incident-Free	----	0.24	0.23
Accidents	----	0.34	----

Table 4-61 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 7.8×10^{-8} LCF.

The highest estimated accident MEI risk is 0.000033 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-61, the total incident-free population risk would be 0.24 LCF for the potentially exposed public, while the corresponding risk would be 0.23 LCF for workers. Thus, there would be an estimated 24 percent chance of incurring one additional LCF among the exposed general public, and a 23 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-61. There is about a seven percent chance that a truck driver or member of the public could die in a traffic accident associated with this Hybrid Alternative. This death would be unrelated to the radioactive nature of the cargo.

4.6 No Action Alternative

Under the No Action Alternative, no foreign research reactor spent nuclear fuel or high-level waste would be accepted into or managed by the United States. The United States would not provide any technical or financial assistance to foreign research reactor operators for the management of their spent nuclear fuel. The United States would rely on the foreign governments' compliance with existing international agreements to control the disposition of foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

Policy Considerations

The No Action Alternative would have a major adverse impact on U.S. nuclear weapons nonproliferation policy. The No Action Alternative would not remove any of the approximately 4.6 metric tons of U.S. origin HEU from international commerce as considered under the proposed action. Under this alternative, the foreign research reactor owners would continue, or may revert back to, use of HEU fuel in their reactors. Countries that can reprocess might send their HEU spent nuclear fuel to be reprocessed and use the separated HEU to produce fresh HEU fuel. In addition, any new research reactors to be built would likely be designed to use HEU fuel. Thus, the No Action Alternative could cause an increase in the number of shipments of weapons-grade nuclear material in transit around the world. It would also damage, perhaps irreparably, the credibility of the RERTR program. Countries that cannot reprocess their research reactor spent nuclear fuel would have to store their fuel. As the spent nuclear fuel ages, it becomes less dangerous to handle (its radioactivity decreases with time), and could possibly become a target of theft and diversion. Hence, the No Action Alternative would undermine the U.S. nuclear weapons nonproliferation policy and the risk of weapons-grade nuclear material being diverted into a nuclear weapons program would increase markedly.

To demonstrate the risk of having reactor owners continue, or revert back to, use of HEU fuel, please see Tables B-3, B-4, and B-5 in Appendix B. These tables list the 104 foreign research reactors whose spent nuclear fuel is included under the proposed action, including 24 reactors that have been converted (fully or partially) or are in various stages of conversion (i.e., ordered, or anticipated to begin converting) from HEU to LEU fuel, and 30 reactors that could be converted, but are not being converted, because the owners of the research reactors are awaiting the outcome of this EIS before they make a decision. Under the No Action Alternative, it is possible that up to 48 foreign research reactor operators could choose to continue or revert back to using HEU fuel in their reactors. These tables also list 23 foreign research reactor operators who possess HEU spent nuclear fuel, even though their reactors are either already shut down or planned to be shutdown for various reasons. This HEU spent nuclear fuel would remain in the foreign research reactor host countries, if the No Action Alternative is selected.

On the other side of the ledger, the benefits obtained from research reactors, described briefly in Section 1.1 of the EIS, would be diminished. Since the No Action Alternative means no U.S. assistance to foreign research reactor operators for managing their spent nuclear fuel, additional research reactors may be forced to shut down, because of lack of funds and/or long term storage capabilities. DOE and the Department of State cannot estimate the number of reactors that would actually be shut down because this would depend on each country's regulations regarding spent nuclear fuel storage. Nevertheless, the medical, industrial and environmental services provided by the shutdown research reactors would be lost. For medical services in particular, foreign research reactors produce radioisotopes used in nuclear medicine in the United States (as discussed in Sections 1.1 and 4.3.1.3 of the EIS). If some of these reactors were forced to shut down, a shortage of medical radioisotopes could occur in the United States. Since the U.S. medical requirements for radioisotopes are not likely to decrease in the near future, alternative sources would have to be found. This could involve an increased level of activity at existing U.S. research reactors or construction of a new reactor in the United States to supply the needed medical radioisotopes, with all the potential environmental impacts of these actions.

Environmental Impacts of Overseas Storage without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved might be greater and the quality of storage technology in some countries might be lower than if the U.S. was involved. Under this option, there would be environmental impacts in foreign countries, but none on U.S. territory, unless some of the material was diverted into nuclear weapons production.

Environmental Impacts of Overseas Reprocessing without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts in foreign countries, but none on U.S. territory, except possibly in cases where some of the material was diverted into nuclear weapons production.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts in foreign countries, but none on U.S. territory. The separated HEU would be more vulnerable to diversion into nuclear weapons production, and the increased reliance on HEU for fuel would increase the number of opportunities for diversion of this weapons grade material.

4.6.1 Overseas Storage Without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved would be greater and the quality of storage technology in some countries might be lower than under the other alternatives. In addition, as the spent nuclear fuel gets older, it becomes less dangerous to handle (its radioactivity decreases with time), and could more easily become a target of theft and diversion.

4.6.2 Overseas Reprocessing Without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts only in foreign nations.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts only in foreign nations.

4.7 Preferred Alternative

As discussed in detail in Section 2.9, the preferred alternative is to accept and manage the foreign research reactor spent nuclear fuel and target material in the United States. Under this alternative, the aluminum-based foreign research reactor spent nuclear fuel and target material would be transported to and managed at the Savannah River Site. The TRIGA foreign research reactor spent nuclear fuel would be transported to and managed at the Idaho National Engineering Laboratory.

The policy considerations, marine transport impacts, port activities impacts, ground transport impacts, and management site impacts of the preferred alternative presented in this section are based on analysis performed for the basic implementation of Management Alternative 1 (Section 4.2), Implementation Alternative 1c (Section 4.3.1.3), Implementation Alternative 6 (Section 4.3.6), and Implementation Alternative 7 (Section 4.3.7).

4.7.1 Policy Considerations

The policy considerations for the preferred alternative are similar to those described in Section 4.2 for Management Alternative 1. A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States' commitment to action such as that embodied in this preferred alternative.

Another crucial consideration associated with the preferred alternative is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

Including target material as part of the preferred alternative maximizes the amount of HEU to be removed from international commerce. This includes all the HEU in the basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons) of heavy metal containing HEU] and all the HEU in the target material in Implementation Subalternative 1c [0.2 metric tons (0.2 tons) of heavy metal containing HEU]. The total amount that would be removed from international commerce is up to 4.8 metric tons (5.3 tons) of heavy metal containing HEU.

DOE's preferred alternative allows for the use of chemical separation under certain circumstances, such as when alternative technologies present higher safety risks, are more costly or are unavailable. If chemical separation is used to process the foreign research reactor spent nuclear fuel, the HEU would be blended down during the separation process to a low-enriched form that is unsuitable for nuclear weapons purposes (the blenddown is also required because the F-Canyon cannot safely process HEU beyond initial dissolution). No plutonium would be separated. Instead, the plutonium would be left in the waste stream with the high-level radioactive chemical separation wastes. In addition, the waste would be handled using technologies that are intended to be used for substantially larger quantities of preexisting wastes (e.g., vitrification of high-level liquid radioactive wastes, grouting for low-level wastes, and incineration for some supernatant).

This potential method of handling the foreign research reactor spent nuclear fuel would be consistent with United States nonproliferation policy, despite the use of chemical separation, because (1) it would reduce the worldwide stockpiles of this nuclear weapons material; (2) no plutonium would be separated; and (3) the chemical separation would not be taking place for either nuclear weapons or nuclear power purposes.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These considerations will be addressed by the independent study described in Section 2.9.

4.7.2 Marine Transport Impacts

The marine transport impacts of the preferred alternative would be similar to those of the basic implementation of Management Alternative 1, with the addition of the target material shipments. As discussed in Section 4.3.1.3 and Appendix B, Section B.1.5, target material would be prepared for transport by changing it into either oxide or calcine form, and both forms might be accepted at some time during the proposed policy period. Even though it requires less marine transport, the oxide form presents a higher radiological risk under accident conditions because its smaller particle size is more easily dispersed in air. Therefore, to be conservative, the analysis of marine and port radiological accidents is based on the assumption that all the target material would be shipped as an oxide. The rest of the marine and port target material transport analysis is based on the assumption of 15 cask shipments, which is the maximum number of marine target material casks. This represents an increase of approximately two percent over the 721 marine cask shipments in the basic implementation of Management Alternative 1.

Marine transport to the West Coast of the United States would be limited to a maximum of approximately 38 casks, which slightly decreases the total number of days the ships would be at sea. Furthermore, DOE would strive to minimize the number of shipments necessary by coordinating shipments from several reactors at a time (i.e., by placing multiple casks [up to 8] on a ship). DOE currently estimates that approximately 5 shipments through the Naval Weapons Station at Concord, California would be necessary.

Impacts of Incident-Free Marine Transport

The highest estimated maximally exposed worker risk due to foreign research reactor spent nuclear fuel is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years (Table 4-2). This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for all of the ships' crews involved in the marine transport of foreign research reactor spent nuclear fuel is about 0.034 LCF, as discussed in Section 4.2.1.2.

Target material contains far less radioactivity than foreign research reactor spent nuclear fuel. Each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 marine casks of target material.

Impacts of Accidents During Marine Transport

The risks associated with accidents at sea are bounded by the risks of the same accidents in ports because humans in the vicinity of accidents at sea are much fewer in number than even the least populated port.

Marine Transport Cumulative Impacts and Mitigation Measures

The marine transport cumulative impacts and mitigation measures for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.1.4 and 4.2.1.5, respectively.

4.7.3 Port Activities Impacts

Although all of the candidate ports of entry presented in Section 3 are acceptable, based on the port selection criteria described in Appendix D, DOE would prefer to use military ports. All aluminum-based foreign research reactor spent nuclear fuel and target material from overseas would arrive at candidate ports on the East Coast of the United States, preferably the Naval Weapons Station at Charleston, South Carolina. Up to approximately 38 casks of TRIGA foreign research reactor spent nuclear fuel would arrive at candidate ports on the West Coast of the United States, preferably the Naval Weapons Station at Concord, California.

Impacts of Incident-Free Port Activities

As shown in Table 4-5, the highest maximally exposed worker risk is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for port workers is about 0.012 LCF, as discussed in Section 4.2.2.3.

As discussed under *Impacts of Incident-Free Marine Transport* above, each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 cask shipments of target material.

Impacts of Accidents During Port Activities

The radiological risks due to port accidents were estimated in the same manner as for the basic implementation (Section 4.2.2.3) and Implementation Alternative 1c (Section 4.3.1.3) of Management Alternative 1. The highest estimated population risk for the entire preferred alternative program is 7.1×10^{-7} LCF. This risk estimate is lower than the earlier alternatives due to the use of military ports in the preferred alternative. These ports are located in areas of low population density, so the number of people potentially affected is much lower. The addition of target material causes a very small incremental increase (3×10^{-9} LCF) in the risk.

Port Activities Cumulative Impacts, Mitigation Measures, and Environmental Justice

The port activities cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.2.4, 4.2.2.5, and 4.2.2.6, respectively.

4.7.4 Ground Transport Impacts

The ground transport impacts were calculated under the assumption that only military ports would be used. DOE has selected military ports close to the management sites (the Charleston NWS in South Carolina and the Concord NWS in California) as the preferred ports of entry.

The risk estimates were maximized by assuming all target material would be oxide for radiological accident calculations and calcine for all other calculations. The calcine form could require up to 125 casks of target material to be shipped overland from Canada.

The preferred points of entry, destinations, and approximate numbers of cask shipments in the preferred alternative are presented in Table 4-62. Other shipment distributions would also be possible.

Table 4-62 Points of Entry, Destinations, and Numbers of Shipments in the Preferred Alternative

	Point of Entry			
Cargo and Destination	*East Coast*	*West Coast*	*Canadian Border*	*Total Cask Shipments*
Aluminum-Based Foreign Research Reactor Spent Nuclear Fuel to the Savannah River Site	559	0	116	675
TRIGA Foreign Research Reactor Spent Nuclear Fuel to the Idaho National Engineering Laboratory	124	38	0	162
Target Material to the Savannah River Site	up to 15	0	up to 125	up to 140
Total Cask Shipments	**up to 698**	**38**	**up to 241**	**up to 977**

Impacts of Incident-Free Ground Transport

The incident-free ground transport of foreign research reactor spent nuclear fuel and target material is estimated to result in a maximum of 0.089 LCF over the entire duration of the program. This is the sum of the estimated number of radiation-related LCF to the public and transportation workers.

The estimated maximum number of radiation-related LCF for transportation workers is 0.022. The estimated maximum number of radiation-related LCF for the general public is 0.067, and the estimated maximum number of non-radiation-related fatalities from vehicular emissions is 0.018.

Impacts of Accidents During Ground Transport

The total ground transport accident population risks for the preferred alternative are estimated to be less than 0.00072 LCF from radiation and 0.052 from traffic collisions.

The maximum foreseeable offsite transportation accident would involve a transportation cask of oxide target material in a suburban population zone under neutral (average) weather conditions, which could expose the MEI to 150 mrem. A similar event involving a transportation cask of spent nuclear fuel could expose the MEI to 2.4 mrem. These events are both in the highest accident severity category. Taking all the possible consequences and frequencies of these accidents into account, and adding the foreign research reactor spent nuclear fuel risks with the target material risks yields the MEI risk of 2.7 x 10^{-11} LCF for the preferred alternative.

Ground Transport Cumulative Impacts, Mitigation Measures, and Environmental Justice

The ground transport cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.3.4, 4.2.3.5, and 4.2.3.7, respectively.

4.7.5 Management Site Impacts

As discussed in Section 2.9, all the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory. The fuel would be received and stored in existing facilities. The environmental impacts of the preferred alternative at the Idaho National Engineering Laboratory can be estimated from the environmental impact analysis presented for the basic implementation of Management Alternative 1 (Section 4.2).

At the Savannah River Site, however, the impacts would vary depending on the specific outcome of the preferred management strategy at the site. Aluminum-based foreign research reactor spent nuclear fuel and target material would be managed at the Savannah River Site. The management of the foreign research reactor spent nuclear fuel is based on the schedule for successful implementation of a new treatment and/or packaging technology. If such a new technology could not be successfully demonstrated by the year 2000, chemical separation of a portion of the foreign research reactor spent nuclear fuel might be implemented. The foreign research reactor spent nuclear fuel and target material that is not chemically separated would be stored in existing facilities at the Savannah River Site until the new technology is operational.

Since the preferred alternative includes the construction and operation of an unspecified treatment and/or packaging technology at the Savannah River Site, the environmental impacts of this alternative at this site cannot be estimated with precision. DOE expects, however, that the radiological and nonradiological health and environmental effects from the construction and operation of facilities that would support a new technology would not exceed those estimated for construction of new dry storage facilities and operation of a conventional chemical separation facility evaluated in Sections 4.2.4.2 and 4.3.6 of this EIS. This expectation is based on the following general principles:

- New facilities would be constructed using current DOE design criteria which have evolved on the basis of increased protection of the public, workers, and the environment.

- The primary source of radiological releases from the chemical separation process is the front end dissolution of the spent nuclear fuel matrix. None of the new technologies considered involves a process that would produce greater releases.

- One of the reasons for the development of a new treatment and/or packaging technology is to reduce the volume and toxic nature of low-level and hazardous waste streams, an issue considered to be a disadvantage of the chemical separation process.

Nonradiological impacts from the construction of facilities that would support the new technology are expected to be typical to those assessed for the construction of new staging and storage facilities assessed for the basic implementation of Management Alternative 1 in Section 4.2.4.2. These include land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air and water quality, ecology, noise, materials and energy consumption, and non-radiological or non-toxic waste production during construction.

The occupational and public health and safety, waste management, and cumulative impacts presented below assume that the implementation of the preferred alternative at the Savannah River Site would result in radiological health effects equal to those presented in Sections 4.3.6 and 4.3.7 of this EIS.

4.7.5.1 Occupational and Public Health and Safety

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 foreign research reactor spent nuclear fuel elements that would be received and managed at the Idaho National Engineering Laboratory under the preferred alternative represent about 22 percent of the total number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the preferred alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Adjusting the figures from Table 4-9 to account for the reduced amount of material in the preferred alternative yields the highest estimated risks for this part of the preferred alternative. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be zero.

The incident-free radiological public health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site.* The highest estimated public MEI risk is 0.0000043 LCF and the highest estimated public population risk is 0.18 LCF.

The maximum of the onsite activities' estimated public incident-free MEI risks is equal to 0.0000043 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in one hundred thousand.

The total of the onsite activities' estimated incident-free population risks to the people who live near both sites is equal to 0.18 LCF. This number means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the two sites due to these incident-free activities.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of the receipts in the preferred alternative is 13 years, the same as that in the basic implementation, the target material alternative, and the chemical separation alternative of Management Alternative 1. Thus, the estimated maximally exposed worker dose is also the same. The highest maximally exposed worker risk is estimated to be 0.026 LCF.

The incident-free worker population risks of the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4.1. Using the same evaluation process yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the preferred alternative is 0.021 LCF.

The incident-free radiological worker health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site*. The highest estimated worker population risk is 0.21 LCF.

The total of the onsite activities' estimated incident-free worker population risks at both sites is 0.23 LCF, which means that there would be an approximately 23 percent chance of one additional LCF among the affected radiation workers at the two sites.

Impacts to the Public of Accidents Onsite

Accident scenarios, frequencies, consequences, and risks for the preferred alternative at the Idaho National Engineering Laboratory are the same as those for the basic implementation of Management Alternative 1. The estimated accident frequencies and consequences are presented in Table 4-20. The highest estimated public MEI/NPAI consequence is 0.000015 LCF and the highest estimated public population consequence is 1.0 LCF. Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory are presented in Table 4-25. Multiplying these figures by the appropriate duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the bounding radionuclide inventory presented in Appendix B, Table B-6 that was used for the evaluations in the basic implementation of Management Alternative 1. The highest estimated accident MEI/NPAI risk for this part of the preferred alternative is 0.0000019 LCF, which is due to a criticality event at an existing wet storage facility. The highest estimated accident population risk for this part of the preferred alternative is 0.016 LCF, which is due to an accidental fuel assembly breach.

The radiological public health impacts due to accidents at the Savannah River Site under the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Impacts of Chemical Separations Accidents at the Savannah River Site*. The highest estimated public MEI risk is 0.000047 LCF and the highest estimated public population risk is 0.43 LCF.

The maximum of the onsite activities' estimated accident public MEI risks is equal to 0.000047 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in ten thousand.

The total of the onsite activities' estimated accident population risks at both sites is equal to 0.45 LCF. This means that there would be an approximately 45 percent chance that one additional LCF would be incurred among the people living near both sites due to accidents during these activities.

4.7.5.2 Waste Management

Implementation of the receipt and storage portions of the preferred alternative would introduce a very small increase in waste generation over current levels at both sites. Baseline site generation of waste is shown in Appendix F, Tables F-23 and F-46 for the Savannah River Site and the Idaho National Engineering Laboratory, respectively. It should be noted that the figures represent storage of more fuel elements, at both sites, than the amounts indicated by the preferred alternative. Implementation of a new technology would produce waste in the amounts presented in Table 4-56.

If the chemical separation portion of the preferred alternative is implemented, this would generate different wastes at the Savannah River Site in place of some of the waste from the new technology. As discussed in Section 4.3.6.6.5, the primary wastes generated during conventional chemical separation and vitrification operations are high-level waste glass in canisters and saltstone. Assuming the chemical separation portion of the preferred alternative could involve up to approximately one-third of the aluminum-based foreign research reactor spent nuclear fuel (6,000 elements), this waste generation would be about one-third of the amount generated under Implementation Alternative 6. Under the preferred alternative, DOE could generate up to approximately 24 high-level waste glass canisters and 1,350 cubic meters (47,700 cubic feet) of saltstone. These wastes would be managed along with much larger quantities of identical wastes in existing facilities at the Savannah River Site.

4.7.5.3 Cumulative Impacts

Cumulative impacts from the implementation of the preferred alternative at both the Idaho National Engineering Laboratory and the Savannah River Site are expected to be lower than those presented for the basic implementation of Management Alternative 1 in Sections 4.2.4.3.1 and 4.2.4.3.2 for the two sites, respectively. At both sites the cumulative impacts from the management of foreign research reactor spent nuclear fuel and impacts from other existing or planned activities or actions at the sites, as presented in Tables 4-29 and 4-30 for Savannah River Site and Idaho National Engineering Laboratory, respectively, including activities not related to the management of spent nuclear fuel, would not challenge or have detrimental effects on the public or environmental resources at the sites.

4.7.5.4 Mitigation Measures

Although environmental impacts at both the Savannah River Site and the Idaho National Engineering Laboratory for the implementation of the preferred alternative would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Such measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Section 4.2.4.6 provides details on these issues.

4.7.5.5 Environmental Justice

The environmental justice conclusions for the management sites discussed in Section 4.2.4.5 for the implementation of Management Alternative 1 are valid for the preferred alternative. As discussed in Section 4.2.4.5, minority or low-income populations living near the Savannah River Site or the Idaho National Engineering Laboratory would not be subjected to any disproportionately high and adverse impacts.

4.7.6 Short Term Uses and Long Term Productivity

The use of land at the Savannah River Site for the potential construction of the new technology facilities would conform with the land use policy at the site. After adoption of an overall strategy for the management of all DOE-owned spent nuclear fuel (including spent nuclear fuel from foreign research reactors), some of the areas may be released for other productive uses.

4.7.7 Irreversible and Irretrievable Commitments of Resources

The operation of existing storage facilities at both sites would involve the consumption of some irretrievable amounts of electrical energy. The potential construction of new technology facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in the construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

4.7.8 Summary of the Impacts of the Preferred Alternative

The principal impacts of the preferred alternative would be occupational and public health and safety impacts. These are presented in Table 4-63 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF due to the preferred alternative. The population risk expresses the estimated number of additional LCF among the entire potentially exposed population.

Table 4-63 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free risk for individual members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to an accident under this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-63, the total incident-free population risk would be 0.25 LCF for the potentially exposed public, while the corresponding risk would be 0.30 LCF for workers. Thus, there would be an estimated 25 percent chance of incurring one additional LCF among the exposed general public, and a 30 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Table 4-63 Maximum Estimated Radiological Health Impacts of the Preferred Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 7.1×10^{-7}	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	7.1×10^{-7}	---
Ground Transport			
Incident-Free	0.00052	0.067	0.022
Accidents	2.7×10^{-11}	0.00072	---
Site Activities			
Incident-Free	0.026	0.18	0.23
Accidents	0.000047	0.45	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.25	0.30
Accidents	----	0.45	----

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-63. There is approximately a five percent chance that a truck driver or member of the public could die in a traffic accident associated with the preferred alternative. This death would be unrelated to the radioactive nature of the cargo.

4.8 Comparison of the Alternatives

This chapter has identified the policy considerations and potential environmental impacts resulting from the proposed action, with all of its various alternatives, and the No Action Alternative. This section provides a comparison of the potential impacts of each alternative, with emphasis on key issues such as the amount of HEU removed from international commerce and risks to workers and the public.

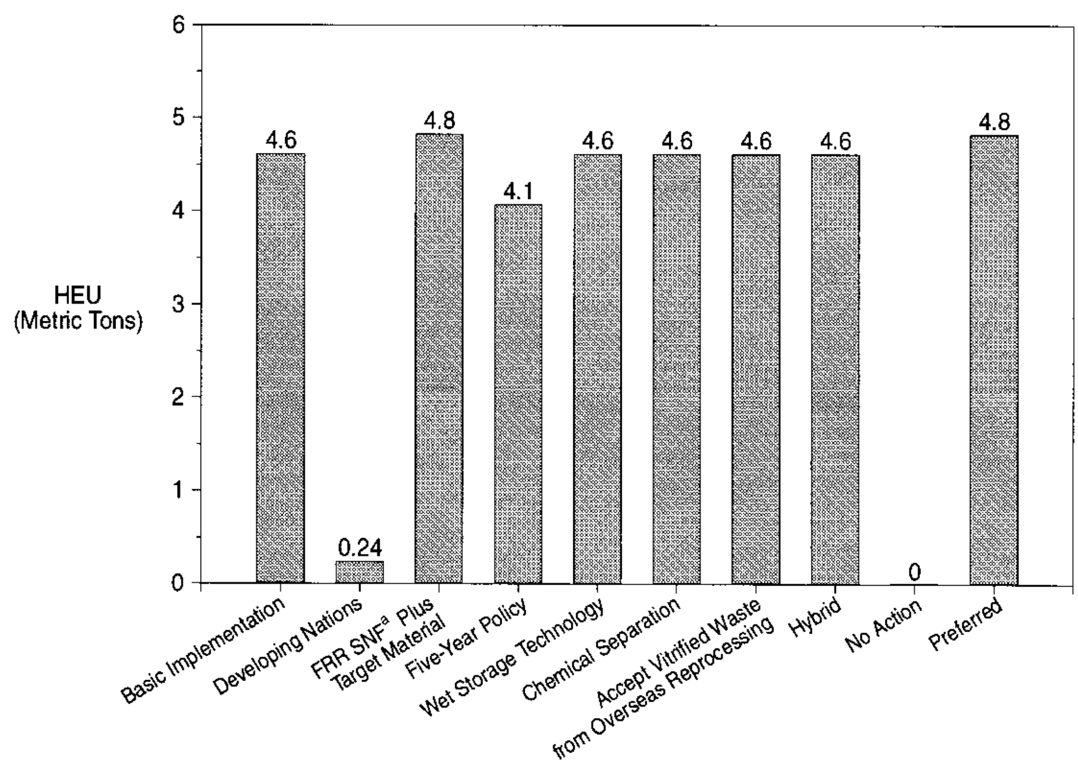
4.8.1 Amount of HEU Removed from International Commerce

The purpose and need for Agency action is driven by the concern that HEU in civilian commerce might be diverted into a nuclear weapons program. Removal of HEU from international civilian commerce will greatly enhance the goals of the U.S. nuclear weapons nonproliferation policy. Figure 4-21 compares the quantities of HEU that would be removed from international civil commerce under the basic implementation of Management Alternative 1, the implementation alternatives, the Hybrid Alternative, the No Action Alternative, and the preferred alternative.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would remove up to an estimated 4.6 metric tons (5.1 tons) of HEU from international commerce. By accepting this weapons-grade material into the United States for storage, the risk of material diversion would be eliminated. For comparison, the United States moved about 0.6 metric tons (0.7 tons) of HEU from Kazakhstan to the United States in November and December 1994 to ensure that it could not be diverted into a nuclear weapons program. The quantity of HEU involved in the basic implementation of Management Alternative 1 is over seven times the amount removed from Kazakhstan. The HEU in foreign research reactor spent nuclear fuel, however, is mixed with fission products, so it would require more sophisticated chemical processing to convert it to uranium metal suitable for use in nuclear weapons.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation of Management Alternative 1 could have an impact on the amount of HEU in international civil commerce. As shown in Figure 4-21, the implementation alternative of accepting spent nuclear fuel only from developing nations would remove up to only about 0.24 metric tons (0.26 tons) of HEU from international commerce. The implementation alternative of accepting target material in addition to the foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1 would remove the most HEU (up to 4.8 metric tons or 5.3 tons) from international commerce. If the acceptance policy lasted for only 5 years, then the amount of HEU involved would be only up to 4.1 metric tons (4.5 tons).

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 could indirectly impact the amount of HEU removed from international commerce depending on whether those financial adjustments influence the amount of foreign research



[a] Foreign Research Reactor Spent Nuclear Fuel

Figure 4-21 Quantities of HEU that Would Be Removed from International Commerce Under Each Alternative

reactor spent nuclear fuel transported to the United States. The final amount of HEU removed from international civil commerce through the application of different financial arrangements cannot be readily determined at this point.

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international commerce, i.e., the action would still remove up to 4.6 metric tons (5.1 tons) of HEU. Similarly, the use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international civil commerce, since the alternative relates to actions within the United States. Implementation by use of near term chemical separation in the United States instead of interim storage would also cause no change in the amount of HEU removed, again because the alternative involves actions in the United States.

Storing foreign research reactor spent nuclear fuel at one or more overseas sites would have a questionable effect on the amount of HEU removed from international commerce. Although this management alternative would provide the United States some limited measure of control over the foreign research reactor spent nuclear fuel, the prevention of material diversion into a nuclear weapons program would not be as fully ensured as if the foreign research reactor spent nuclear fuel was accepted into the United States. This alternative would leave HEU stockpiled around the world.

The implementation alternative of overseas reprocessing would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

Hybrid Alternative: The Hybrid Alternative chosen for analysis would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

No Action Alternative: Under this alternative, the United States would rely solely on the foreign governments' compliance with international agreements to control the foreign research reactor spent nuclear fuel. A policy of no action by DOE and the Department of State runs counter to U.S. nuclear weapons nonproliferation policy by causing continued reliance on HEU, thus not realizing the goal of eliminating civil commerce in HEU.

Preferred Alternative: The preferred alternative would remove the same amount of HEU (up to 4.8 metric tons or 5.3 tons) from international commerce as would Implementation Alternative 1c of Management Alternative 1. This amount is higher than for the other alternatives.

4.8.2 Radiological Risk to Individuals

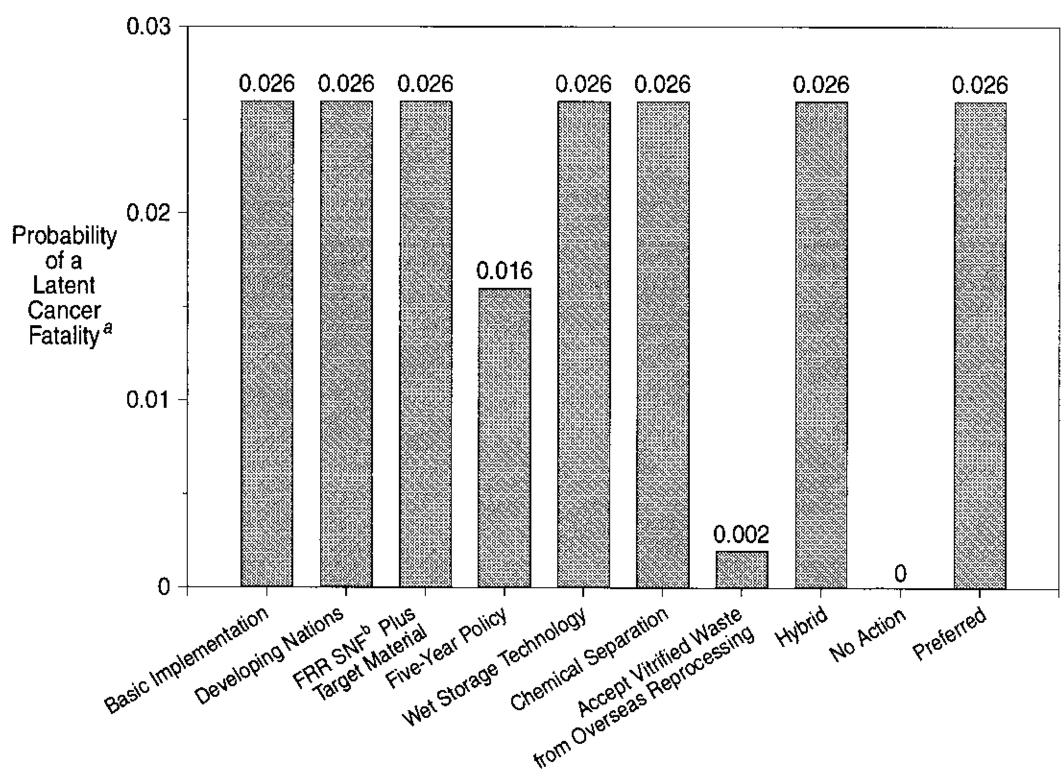
A maximally exposed worker or an MEI in the public is a hypothetical individual who records the highest possible exposure to radiation in a given situation, and the associated risks are different depending on the alternative considered. Figures 4-22 and 4-23 present comparisons of the estimated radiological risk to the maximally exposed worker and to the MEI under each alternative for incident-free and accident conditions, respectively. Alternatives involving the smallest number of cask shipments into the United States would produce the lowest individual risks. There would be no maximally exposed worker risk or MEI risk in the United States under the No Action Alternative.

The incident-free maximally exposed worker risk estimates are driven by the assumption that a radiation worker would receive the maximum radiation dose allowed by law for every year that foreign research reactor spent nuclear fuel is accepted. This risk depends only on the duration of the action, not on the number of casks or elements. Thus, the Five-Year Acceptance Alternative would present lower risk than the alternatives which last for 13 years.

The accident MEI risk estimates are dominated by onsite accident scenarios. This is because during marine transport, port activities, and ground transport, the foreign research reactor spent nuclear fuel would be inside transportation casks. During onsite activities, while spent nuclear fuel is outside of transportation casks, the probability of an incident that could release radioactive material is higher. The highest estimated accident MEI risk in the public is 0.00015 LCF, which means that this hypothetical individual's increased chance of incurring an LCF would be less than two in ten thousand.

4.8.3 Radiological Risk to Exposed Populations

Population risk is the risk of additional latent cancers occurring among people (both public and workers) who would be exposed to radiation. Risks vary with the alternative considered. Figures 4-24 and 4-25 present comparisons of the estimated incident-free radiological risks to the public and worker populations under each alternative. Alternatives involving the smallest number of cask shipments into the United



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-22 Maximum Estimated Incident-Free Radiological Risk to the Maximally Exposed Worker Under Each Alternative

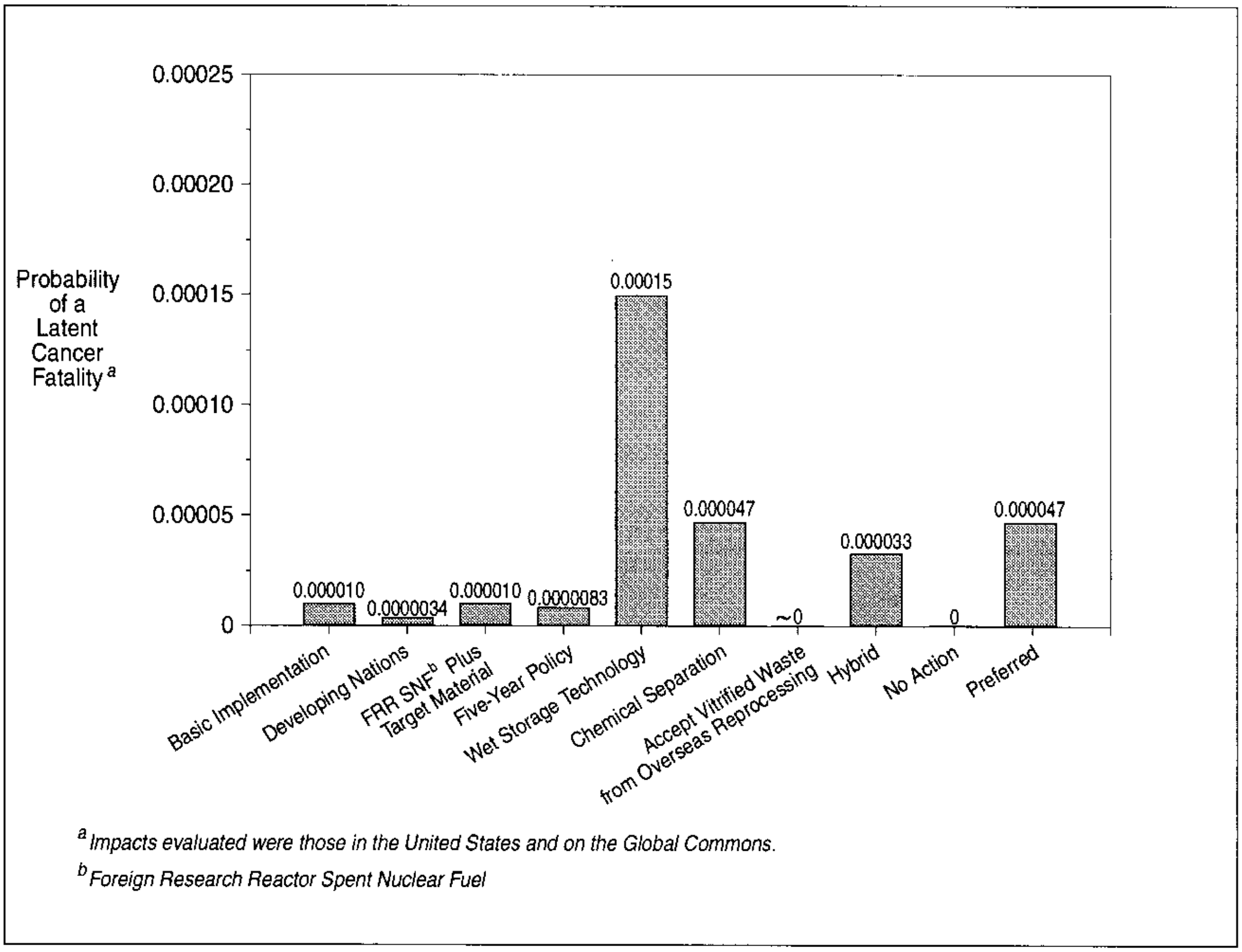
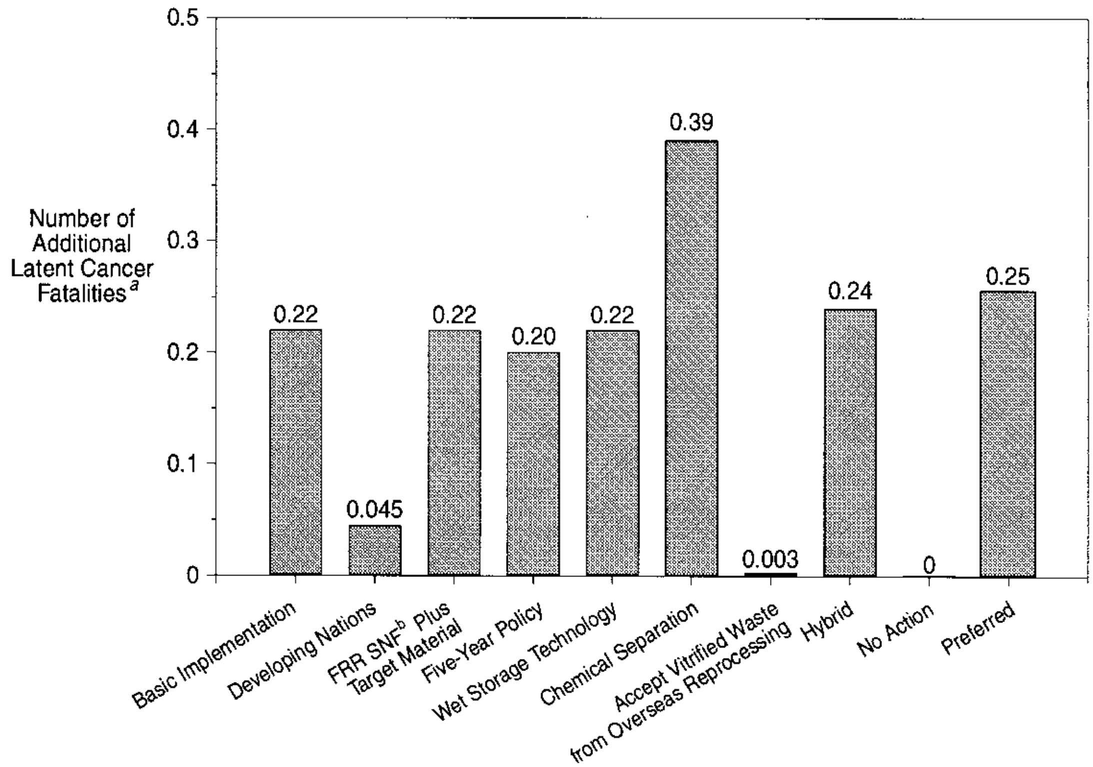


Figure 4-23 Maximum Estimated Accident Radiological Risk to the MEI in the Public Under Each Alternative

States would produce the lowest population risks. The chemical separation, overseas reprocessing, and preferred alternative are the alternatives in which the waste would be conditioned for disposal. Under the other alternatives, some form of processing may be required at some time in the future before disposal. There would be no population risk in the United States under the No Action Alternative. Under all the alternatives the estimated incident-free public and worker population risks would result in less than one-half additional LCF among each population group.

Figure 4-26 presents a comparison of the estimated accident radiological population risks to the public under each alternative. Those alternatives involving some form of processing in the United States would present the largest accident risks, but these risks would occur in the near term. Under the other alternatives, some form of processing may be required at some time in the future before disposal. Under all the alternatives, the estimated accident public population risks would result in less than one-half additional LCF.



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-24 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative

4.8.4 Nonradiological Risks

The transport of foreign research reactor spent nuclear fuel from the ports to the sites would involve some risk of death due to traffic accidents for both the truck drivers and the public. Figure 4-27 presents a comparison of the estimated traffic accident risk to both the drivers and public combined under each alternative. Estimates include the risks associated with transporting the empty casks back to the ports.

Results are directly proportional to the number of highway miles over which casks would be transported under each alternative. The basic implementation of Management Alternative 1 and four of the implementation alternatives would have essentially the same risk, while the Developing Nations Subalternative and the Hybrid Alternative would have lower traffic accident risks.

Under the subalternative of accepting vitrified waste from overseas reprocessing, an estimated eight cask shipments would be accepted in the United States, so the traffic accident risk would be extremely low. There would be no population risk in the United States under the other overseas subalternative, as well as the No Action Alternative.

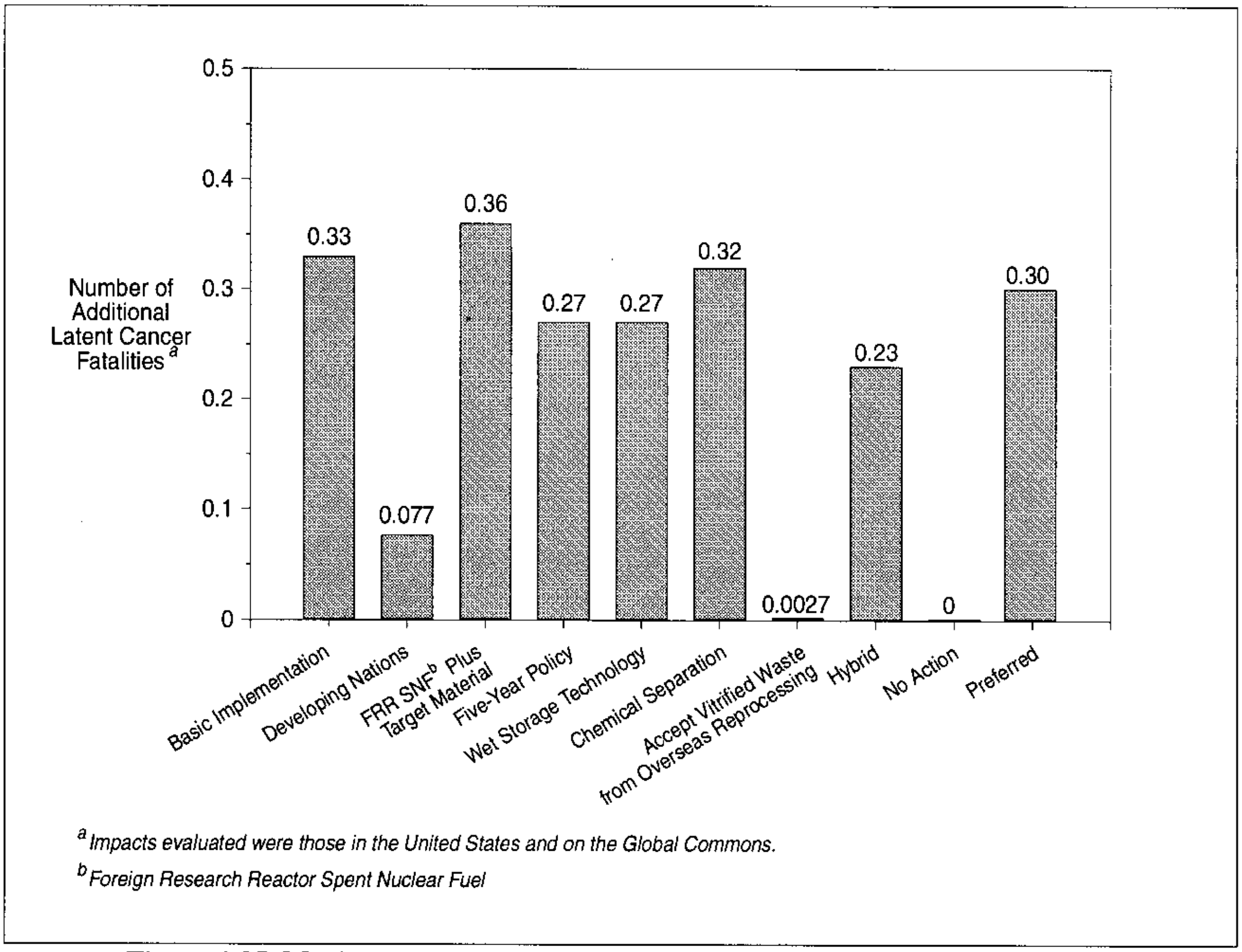


Figure 4-25 Maximum Estimated Incident-Free Radiological Population Risk to Workers Under Each Alternative

The traffic accident risk is also relatively low under the preferred alternative because all the cask shipments of aluminum-based foreign research reactor spent nuclear fuel would go through an east coast port or ports to the Savannah River Site. This effectively minimizes the ground transport risk by minimizing the number of highway miles required.

4.8.5 Land Use

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major land use issues at any of the potential foreign research reactor spent nuclear fuel management sites. If additional storage space were required for the foreign research reactor spent nuclear fuel, the space would be built on DOE-owned lands, inside the boundaries of the DOE management sites.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amount identified in the basic implementation of Management Alternative 1 would not cause land use issues, even though storage needs may vary due to the United States receiving a larger (if target material is accepted in addition to spent nuclear fuel) or smaller (e.g., from developing nations only)

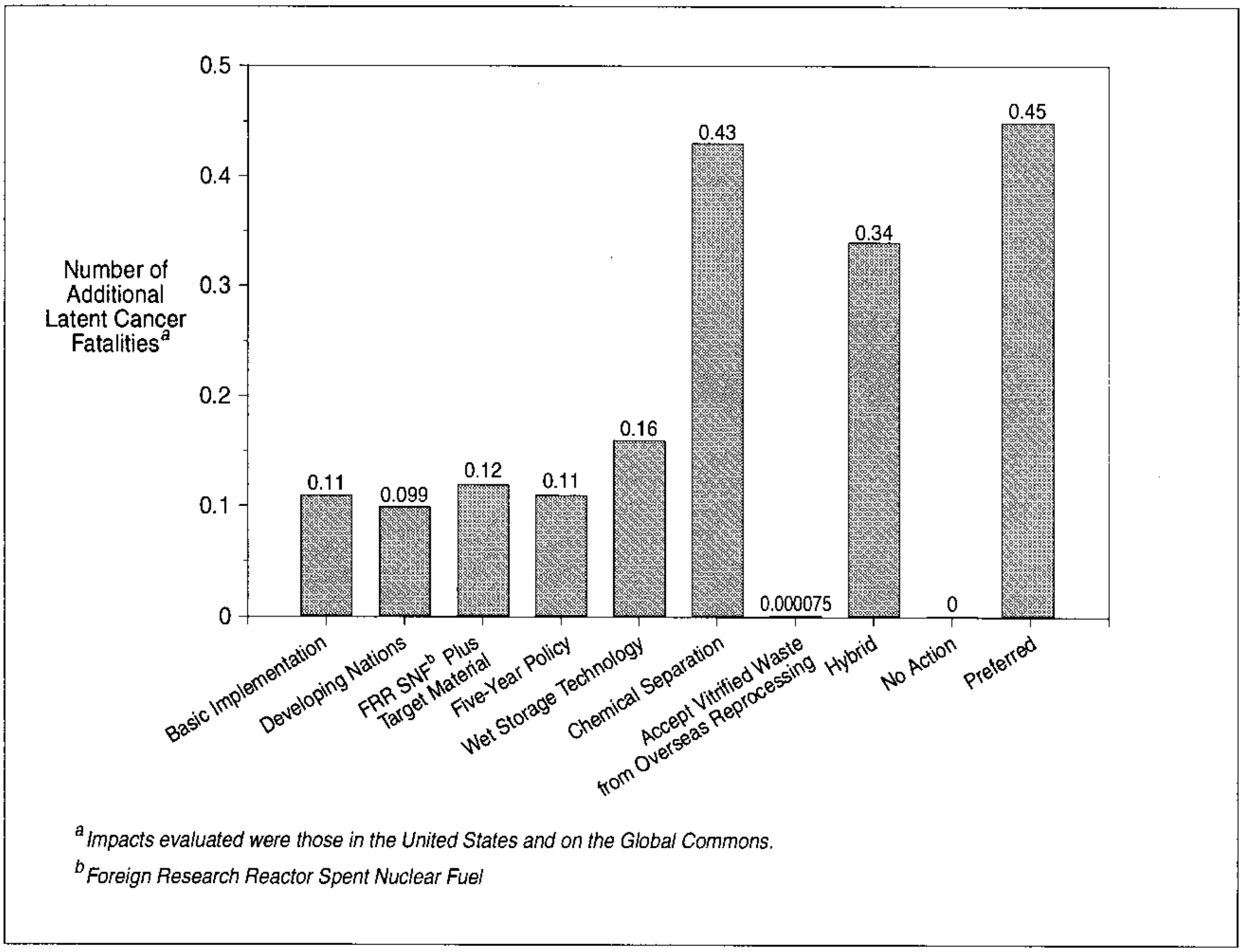


Figure 4-26 Maximum Estimated Accident Radiological Population Risk to the General Public Under Each Alternative

amount of material than identified in the basic implementation of Management Alternative 1. As mentioned above, additional storage space, if required, would be created on DOE-owned land, creating no outside land use issues.

Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the time periods identified in the basic implementation of Management Alternative 1 would not cause any land use issues as the timeframe would not necessarily change the amount of foreign research reactor spent nuclear fuel received by the United States. If a policy of 5 years of acceptance was instituted, less spent nuclear fuel would be received by the United States, and if an indefinite HEU/10-year LEU policy were to be adopted, storage space would be created on DOE management sites, causing no issues in relation to outside lands.

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 would have no impact on land use, as this alternative would have no effect on lands not owned by DOE.

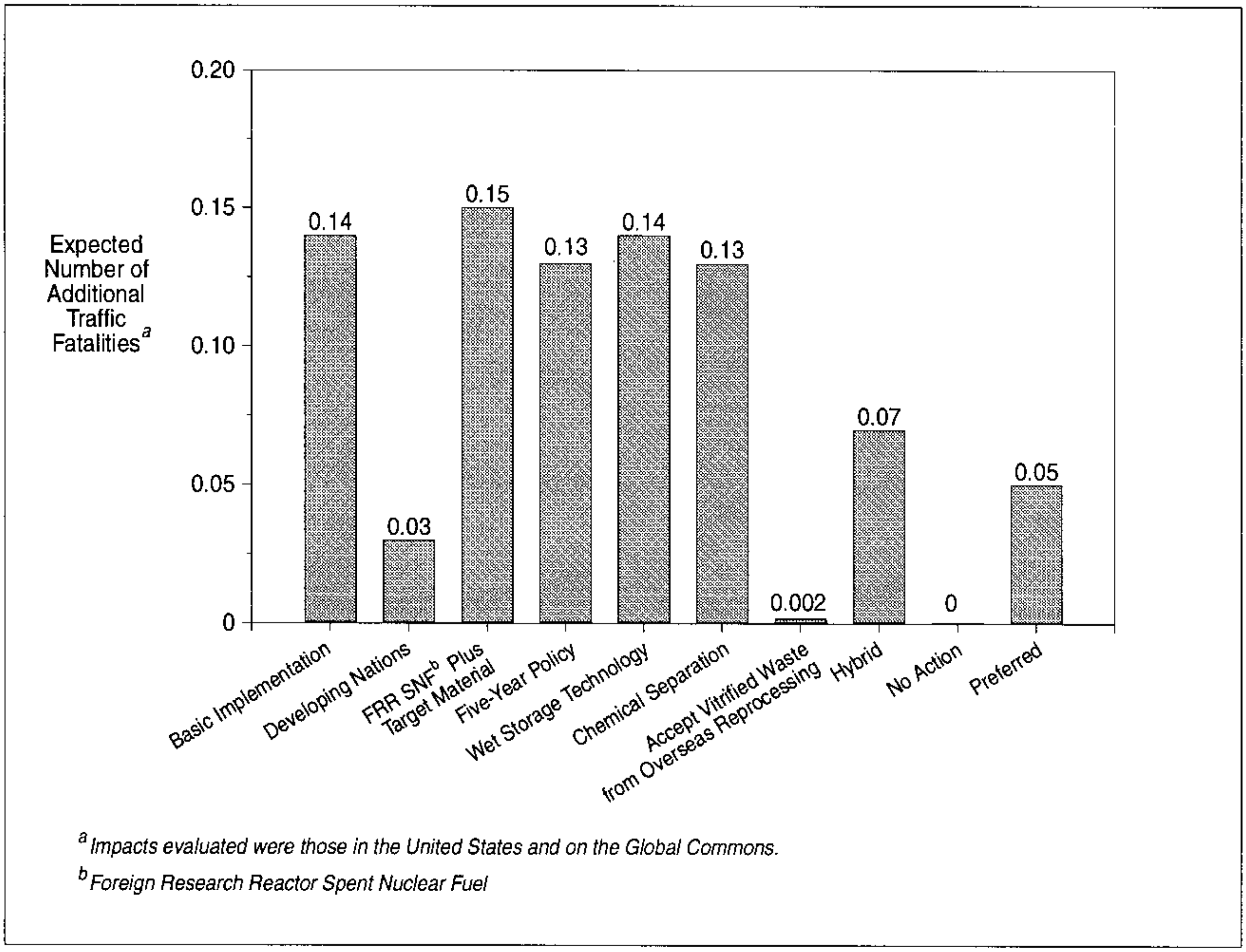


Figure 4-27 Maximum Estimated Traffic Accident Risk Under Each Alternative

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would cause no land use issues, as it would have no effect on the storage needs or the amount of foreign research reactor spent nuclear fuel received by the United States.

Use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would cause no land use issues, as the storage facilities (wet or dry) would be on DOE-owned land, and would have no effect on outside (non-DOE-owned) lands. If DOE decides to purchase the BNFP facility for interim wet storage, however, this would require adding some land to the Savannah River Site.

Implementation by use of near term chemical separation in the United States instead of interim storage would have no impact on land use, as the separation would be performed on DOE-owned land, with no effect on outside (non-DOE-owned) lands.

Similarly, there would be no land use concerns under either of the overseas subalternatives or the Hybrid Alternative presented in this EIS. A policy of no action (the No Action Alternative) regarding foreign research reactor spent nuclear fuel would cause no land use issues in the United States.

Land use for construction under the preferred alternative would be similar to the land use for construction under the basic implementation of Management Alternative 1.

4.8.6 Cultural Resources

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major impact to the cultural resources of the management sites being considered for the storage of the foreign research reactor spent nuclear fuel. Although the sites have not been evaluated and audited for cultural resources, surveys would be completed prior to any construction or other activity that would potentially disturb these areas. Areas of cultural or historical significance are protected by laws and acts (e.g., Native American Grave and Repatriation Act, National Historic Preservation Act, etc.), and the basic implementation of Management Alternative 1 is not likely to have an impact on areas of cultural or historical significance.

Implementation Alternatives: Since the safety of areas of cultural or historic significance is protected under the basic implementation of Management Alternative 1, these areas would not be impacted by any of the various implementation alternatives, the Hybrid Alternative, or the preferred alternative.

The overseas subalternatives would have no impact on cultural resources, as these subalternatives involve no use of DOE management sites. Similarly, the No Action Alternative would have no impact for the same reason.

4.8.7 Air Quality

While all possible precautions and safeguards would be utilized in an effort to conserve air quality, it would be impacted by U.S. acceptance of foreign research reactor spent nuclear fuel.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not be expected to have major impacts on air quality, and projected emissions from foreign research reactor spent nuclear fuel storage at management sites would not violate Federal or State standards. Dust from construction activities could be controlled with standard techniques. Particulate emissions could have temporary effects on localized visibility, but would not adversely affect Federal or State attainment standards.

Implementation Alternatives: Air quality would be most affected under Implementation Alternative 6, the Hybrid Alternative, or the preferred alternative, all of which involve the use of some form of processing in the United States. Chemical separation would yield a higher effect on air quality than any of the other implementation alternatives.

Since the two overseas subalternatives deal strictly with spent nuclear fuel management overseas, and the No Action Alternative involves no action on the part of DOE or the Department of State, air quality in the United States would not be affected under these alternatives.

4.9 Costs

The costs of implementing various scenarios of the proposed action, including the preferred alternative, plus disposal are presented in this section. Additional details pertaining to costs are provided in Appendix F, Section F.7. For the purpose of the cost analysis, the alternatives described in Section 2.1 were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995c) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. DOE selected six scenarios, including the preferred alternative, for cost analysis. The costs of disposal were estimated for each scenario and are included in the analysis. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 that would affect the cost to the United States.

All costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. For the preferred alternative, however, a wide range of costs is presented because of the uncertainty associated with the new technology development program. An example of an item covered by "other cost factors" would be the cost growth caused by adverse weather that extends the time required to make shipments of the foreign research reactor spent nuclear fuel. The costs are shown as net present values in a consistent accounting framework.

4.9.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, and are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The six cost scenarios are:

1. *Management Alternative 1 (Storage)* — Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site in new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory in existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* — Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Target Material* — Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

4. *Management Alternative 2* — Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

5. *Management Alternative 3* — Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

6. *Preferred Alternative* - Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 3 or 2 and 3.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

4.9.2 Minimum Program Costs

Table 4-64 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. Costs to manage target material (Scenario 3) could be added to the costs of Scenarios 1, 2, 4, and 5 to produce a minimum program cost. Costs to manage target material are included in the preferred alternative (Scenario 6).

Table 4-64 Minimum Program Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Target Material	35
4. Management Alternative 2	1,250
5. Management Alternative 3	675
6. Preferred Alternative[b]	625-950

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

The schedule for activities in Europe under Scenario 5 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13 year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States.

Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States. These net present values imply that all funds required to pay for the program over its 40-year life are received and placed in a trust fund accruing interest at a 4.9 percent real rate of return. This rate of return is required by the Office of Management and Budget for the year ending February, 1996.

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 6 (preferred alternative) is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, the shipping costs in Scenario 6 include the assumption that only 38 cask shipments would be accepted on the West Coast.

4.9.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table 4-64. Table 4-65 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence.

Table 4-65 Other Cost Factors (Net Present Value, Millions of 1996 Dollars in 1996)

	Cost Factors				
Scenario	*Systems Integration & Logistics*	*Component Risks*	*Non-program Risks*	*3% Discount Rate*	*Range*
1. Management Alternative 1 (Storage)	100	75	35	175	385
2. Management Alternative 1 (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Target Material	5	5	0	25	35
4. Management Alternative 2	100	±500	1000	250	350-1850
5. Management Alternative 3	100	±10	150	75	315-335
6. Preferred Alternative[b,c]	100	75	35	225	435

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *Includes target material*

[c] *It is assumed that risk factors are the same as Management Alternative 1 (Storage)*

The other cost factors summarized in Table 4-65 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, 13 years of possible receipts, dozens of foreign ports, up to ten domestic ports, two U.S. management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in several areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the comprehensiveness of the acceptance criteria for aluminum-clad spent nuclear fuel characterization for dry storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, (e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory). For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the current discount rate required by the Office of Management and Budget for the year ending February, 1996 (4.9 percent real) will be reduced to a more historically representative level (e.g., 3 percent) in some future annual update. The base case costs for management outside the United States are discounted at a 3 percent rate. The use of a high discount rate is particularly risky because 1) revenues are likely to be fixed (in $/kgTM) early in the program while expenses are variable and uncertain, and 2) revenues received from the reactor operators during the 1996 through 2008 shipping period will almost certainly exceed the costs of management activities during that period. Mathematically, the excess revenues are placed in a trust fund that compounds interest at the discount rate. If the discount rate exceeds the rate at which funds are actually likely to compound, then outyear program costs (e.g., disposal) could not be met from the principal and accrued interest in the trust fund. A reduction in the discount rate from 4.9 percent to 3.0 percent has a larger impact on the program than any of the technical or systems integration risks.

4.9.4 Potential Total Costs

Table 4-66 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table 4-66 Potential Total Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Minimum Program Cost*	*Other Cost Factors (Technical)*	*Other Cost Factors (Discount Rate)*	*Potential Total Cost, No Escalation*	*1% Real Escalation, Cumulative*
1. Management Alternative 1 (Storage)	725/775[a],	210	175	~1,100	+11%
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625	85-145	125	~900	+9%
4. Management Alternative 2	1,250	600-1,600	250	2,100-3,100	+13%
5. Management Alternative 3[c]	675	225-275	75	~1000	+9%
6. Preferred Alternative[b]	625-950	210	225	~1,050-1,400	+10%-11%

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

[c] *The total cost risk to the United States is less than 1/2 the total cost risk because a large portion of the*

Table 4-66 shows that the net present value of the potential total costs of implementing the program completely in the United States, including an estimate of program risks but excluding escalation, range from about $900 million for Scenario 2, to $1.4 billion for Scenario 6. Scenario 5 has similar total program costs as Scenario 2 but higher risks for geologic disposal.

In Scenario 4, costs for storing foreign research reactor spent nuclear fuel overseas are highly speculative. In addition, the overseas storage costs are always higher than the more centralized management alternatives because of the extremely high cost of safely and securely managing and disposing of small quantities of spent nuclear fuel in dozens of countries.

The program costs presented in Tables 4-64, 4-65, and 4-66 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and explicitly or implicitly placed in a trust fund. If payments into the trust fund are deferred, then they must be larger than if they had been received on January 1, 1996. For example, if payments are made in 13 equal annual installments every December 31 over the 1996 through 2008 shipping and receiving period, then the constant-dollar payments must increase by 37 percent. A composite of payment schedules, e.g., 13 years for high-income-economy country reactor operators and pay-as-you-go (for the United States) for all other costs, including other-than-high-income-economy country costs, has the effect of increasing the required constant-dollar payments by as much as 25 to 50 percent.

4.9.5 Cost to the United States

The cost of the proposed policy to the United States depends directly on the type of financing arrangement that DOE would adopt in implementing the policy and the discount rate at which revenues from reactor operators accrue interest. Alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. Briefly, the financing arrangements considered are:

1. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *competitive* fee to high-income-economy countries. This is the financial arrangement in the preferred alternative.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *full-cost-recovery* fee to high-income-economy countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive* versus the *full-cost-recovery* fee. The U.S. cost under the second arrangement (*no* fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table 4-67 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except Scenario 3) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 4 or 5 would increase its costs by 3 to 4 percent. Fees of \$2,000/kgTM, \$5,000/kgTM, \$7,500/kgTM, and \$10,000/kgTM, including a pass-through of shipping charges (all levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation.

The cost to the United States is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the other-than-high-income-economy countries, including shipping, and 2) the difference between the revenues received for management of high-income-economy country foreign research reactor spent nuclear fuel and the total program cost of managing high-income-economy country foreign research reactor spent nuclear fuel, including shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table 4-67 Costs to the United States for Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized over 1996-2008 Period)

Scenario[a]	*Full-Cost Recovery*[b]	*Levelized Shipping Fee $/kgTM*	*Levelized Management Fee (excluding shipping) $/kgTM*	*Net Present Value For Levelized Fee*[c] *(developed countries only)*				*No Fee*[d]	*Total (excluding shipping)*
				$2,000/ kgTM	*$5,000/ kgTM*	*$7,500/ kgTM*	*$10,000/ kgTM*	*Developed Countries*	
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
4. Management Alternative 2[f]	500+							1,250 +	1,750+
5. Management Alternative 3[g]	85	1,500	6,000	225	75	(50)	(175)	300	375
6. Preferred Alternative[e]	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from other-than-high-income-economy countries: 15,000 kgTM; from high-income-economy countries: 100,000kgTM; to Dounreay in Scenario 5: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM, essentially all from high-income-economy countries.*

[b] *Full-cost recovery from high-income-economy countries only. The United States bears the costs of the other-than-high-income-economy countries in these cases.*

[c] *Payable in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States.*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 and 2 is $140 Million. The net present value of shipping to the U.S. only in Scenario 5 is $90 Million. The net present value of shipping in Scenario 6 is $160 million. Adding shipping to the net present value for Scenario 2 and Scenario 5 shows that the total program costs for Scenario 5 are slightly lower.*

[e] *Includes target material*

[f] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Scenario 4 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on their income-economy status, commercial nuclear power infrastructure, and other factors.*

[g] *Revenues paid to the United States include pass-through of shipping charges. Costs to the United States for management in Europe include the cost of blending down the HEU to LEU ($20 million).*

Table 4-67 shows that for minimum discounted program costs and fees charged to high-income-economy country reactor operators levelized over 13 years, costs to the United States for the scenarios could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the other-than-high-income-economy countries (including shipping) adds roughly $100 million more to the cost borne by the United States.

If fees in the $2,000 to $10,000 per kgTM range are established and charged over 13 years, the costs to the United States would be as estimated in Table 4-67 plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table 4-66 shows that technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are uncertain but could be in the same range.

4.10 Foreign Research Reactor Spent Nuclear Fuel Risks and Common Risks

This section compares foreign research reactor spent nuclear fuel program risks to those of common activities, such as smoking, flying, receiving a medical X-ray, and so forth.

4.10.1 Risks in the Proposed Action

Preceding sections in Chapter 4 evaluated the risks from radiological and nonradiological activities and accidents in four segments: marine transport, port activities, ground transport, and site activities.

The highest estimated accident MEI risk to the general public from any of the foreign research reactor spent nuclear fuel implementation alternatives is 0.00015 LCF, as shown earlier in Figure 4-23. This would be an individual who lives at the Oak Ridge Reservation boundary under Implementation Alternative 5, Wet Storage Technology for New Construction. This hypothetical individual's chance of incurring a fatal cancer would be increased by less than two in ten thousand.

The highest estimated incident-free population risk to the general public living near any of the DOE management sites from any of the implementation alternatives is less than one-half LCF, as shown earlier in Figure 4-24. This risk occurs under Implementation Alternative 6, Near Term Chemical Separation in the United States, at the Savannah River Site. This risk would be spread among the roughly 600,000 people who live within 80 km (50 miles) of the Savannah River Site, so the average risk among these people would be less than one in a million.

The population risk to the general public due to radiation exposure during ground transport could be as high as 0.22 LCF, as discussed earlier under several of the implementation alternatives to Management Alternative 1.

Nonradiological fatalities are also unlikely. As a practical matter, the only source of nonradiological fatalities to the public is through a traffic accident with a truck or a train. Since truck or train shipments are about 100 or fewer per year, the likelihood of a crash is not high.

4.10.2 Common Radiological Risks

Table 4-68 presents several typical sources of exposure to radiation from everyday life (DOE, 1993e). The average person in the United States receives about 300 mrem each year from natural sources of radiation and about another 50 mrem from manmade sources of radiation. For example, the largest dose listed in Table 4-68 is the 200 mrem/yr from exposure to naturally-occurring radon gas. This is twice the

100 mrem/yr regulatory limit that would apply to marine workers, port workers, and truck drivers under the proposed action. It is also much higher than the dose any member of the general public would be likely to receive.

Table 4-68 Typical Sources of Radiation, Exposures, and Risks

Source	*Dose Rate (mrem/yr)*	*Risk (LCF/yr)*
Radon	200	0.0001
Internal	39	0.000020
Diagnostic X-rays	39	0.000020
Soil, rocks	28	0.000014
Cosmic rays	27	0.000014
Nuclear medicine	14	0.000007
Nuclear fuel cycle	less than 1	less than 5×10^{-7}
Fallout	less than 0.01	less than 5×10^{-9}

There are also large variations in radiation dose to which people are routinely exposed. For example, people who live at high altitudes receive more radiation dose than people who live at sea level. People who live or work in brick, granite, or marble buildings receive more radiation dose than people who live or work in wooden structures. People who live in well-insulated houses receive more radiation dose from trapped radon gas than people who live in well-ventilated houses. Taking all the various factors into account, the annual U.S. dose from background radiation can easily range from 100 mrem for people who live in well-ventilated wooden houses on sandy soil at sea level to about 1000 mrem for people who live in well-insulated houses in the Denver area (de Planque, 1994). Thus, in addition to the average annual radiation dose, routine variations in annual radiation dose are also much larger than the dose any member of the general public would be likely to receive under the proposed action.

4.10.3 Risks from Common Activities

Every activity carries some risk. Table 4-69 shows risks estimated to increase an individual's chance of death in any year by one in one million (Slovic, 1986). For example, a single airline flight across the United States would increase each passenger's radiation dose by about 4 mrem (de Planque, 1994). Most of these voluntary activities would not be considered unusually risky actions, and they can be compared to the risks presented earlier in this chapter for perspective.

Table 4-69 Risks Estimated to Increase Chance of Death in Any Year by One Chance in a Million

Activity	*Cause of Death*
Smoking 1.4 cigarettes	Cancer; heart disease
Living 2 days in New York or Boston	Air pollution
Traveling 16 km (10 mi) by bicycle	Accident
Flying 1,600 km (1,000 mi) by jet	Accident
Living 2 months in Denver on vacation from New York	Cancer caused by cosmic radiation
One chest X-ray taken in a good hospital	Cancer caused by radiation
Drinking 30 12-oz cans of diet soda	Cancer caused by saccharin